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Status of maritime gas cooled reactors.

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STATUS OF MARITIME GAS COOLED REACTORS

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I. INTRODUCTION

With the advent of large surface effect ships and other large air cushion vehicles, there is renewed interest in the gas cooled reactor for maritime use. Studies have shown the economic value of a low weight to horsepower ratio power plant in sizes ranging from 50,000 shaft horsepower (SHP) to 450,000 SHP.[1,2] It has been indicated in these studies that the most probable nuclear plant to achieve the required low weight to horsepower ratio (5-10 lb/SHP) is the gas cooled reactor, but much development must be done. What then of the development already done? What is the present status of the maritime gas cooled reactor?

This paper will review the present status of gas cooled reactors and maritime reactors in general and then specifically point out the past development of the maritime use of gas cooled reactors. Other mobile gas cooled reactor systems will be reviewed to present related development. Finally a comparison will be made, with general conclusions as to where we stand and which direction we are going. An appendix is included indicating the applicable thermodynamic cycles, specifically the steam cycle and the gas turbine cycle.

II. EXISTING GAS COOLED REACTORS

MAGNOX

The gas cooled reactor has played an historically important role in the development of nuclear energy. The so-called "Daniels Pile" of 1946, which probably reflected the performance of the still earlier air-cooled X-10 Pile at Oak Ridge in 1944, was among the very first reactors seriously proposed for the production of power. The gas cooled reactor is presently the cornerstone of the British nuclear power program. The MAGNOX type of gas cooled reactor, typified by the Calder Hall reactor, has generated, in the period 1956 through 1972, more electric power (over 200 million MWh) than all other nuclear power stations combined.[3,4] Although the power plants of the Calder Hall type are no longer as economically competitive as new designs, their reliability still indicates the viability of the design.

This reactor is a graphite moderated, carbon dioxide cooled reactor fueled with metallic natural uranium clad with Magnox (a magnesium alloy). The fuel elements are massive and the specific power and power density are consequently low. The primary coolant pressures range from 8 atm in the early reactors to 30 atm in the later reactors of this type, typified by the Wylfa reactor. The primary coolant outlet temperatures also range from 345°C to 410°C respectively.

As in all operating central power stations using gas cooled reactors, the primary coolant passes through a heat exchanger to produce steam. The steam is then used to convert the energy to electricity

using the steam cycle in one of its forms as indicated in Appendix A. A general simplified schematic applicable to those reactors utilizing the steam cycle is shown in Figure 2-1. Further detailed information on all the reactors discussed is contained in the tables in Chapter 6.

AGR

The next major central power station development was the British advanced gas cooled reactor (AGR).[3,4,5] The AGR is capable of developing considerably higher gas temperatures and much higher power density along with extended fuel life, and therefore much higher efficiencies - up to 41 percent in the newest plant. The fuel consists of slightly enriched uranium dioxide pellets clad with stainless steel. The coolant is carbon dioxide at pressures of 20-42 atm but the outlet temperatures are from 460°C for Windscale to 650°C for the newer AGR reactors such as Hunterston B. Another major innovation of the newer AGR reactors is the integrated structural design with pressure vessels of prestressed concrete. In fact, the newest MAGNOX reactors also use the prestressed concrete reactor vessel, as do all recently designed gas cooled central station reactors.

HTGR

A concurrent development has been the high temperature gas reactor (HTGR). A 40 MWe prototype has been constructed at Peach Bottom.[6,7] In this reactor the fuel is 93 percent enriched uranium and thorium carbide in the form of small spheres 100-500 μ in diameter clad with approximately 55μ of pyrolytic carbon. These small spheres,

along with similar fertile particles of higher thorium carbide content, are embedded in a matrix of graphite which acts as the moderator. Helium is the primary coolant and it is divided into two streams in the reactor. The larger stream flows around the fuel element and cools it and the smaller stream goes through the fuel element itself and purges it of the fission products which are released from the small coated particles. Therefore, the helium becomes contaminated and must be continuously cleansed of fission products. The helium operates at 24 atm between temperatures of 344°C and 728°C. The hot helium gas produces steam as in Figure 2-1 at 538°C and 100 atm which gives an overall plant efficiency of 35 percent, primarily as a result of the higher temperature.

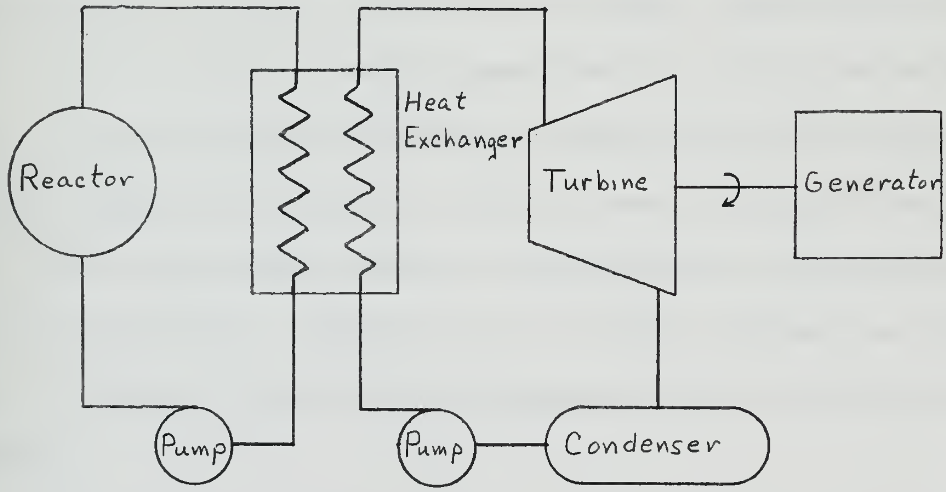

Figure 2-1

Simplified Schematic of Power Conversion
using the Steam Cycle

The British DRAGON reactor experiment[8,9] is a 20 MWth HTGR similar to the Peach Bottom prototype, but with no electrical generating capability. The operating characteristics are nearly identical, but the purpose is the testing of ceramic coated fuel particles. Several different types of coated particles have been tested in the DRAGON reactor since 1966 indicating that more than one layer of pyrolytic carbon can retain the vast majority of fission products released during operation. This means that the purging channels as utilized in Peach Bottom are not necessary.

The 330 MWe HTGR being constructed at Fort St. Vrain, therefore, does not use purge channels.[10,11] Also the fuel elements have been changed from the long spindle type used in Peach Bottom to a short hexagonal block of graphite containing the coated fuel particles and drilled with several coolant and control rod passages. The 200μ fuel particles are coated with a three layer coating. The inner layer consists of 50μ of porous pyrolytic carbon which is called the buffer layer. It is followed by a thin (15μ) layer of silicon carbide which is highly impervious to metallic fission products deposited outside the buffer layer. The final layer is a 50μ deposition of high density isotropic pyrocarbon. Control rods are of boron carbide clad with steel. As with the later AGR, the pressure vessel is prestressed concrete.

The primary coolant is helium at 47 atm operating between temperatures of 400°C and 775°C, producing steam at 538°C and 163 atm. The general schematic is similar to Figure 2-1 however the heat exchanger is integral within the prestressed concrete reactor vessel

and a reheat cycle with two turbines is used (See Appendix A). The overall plant efficiency is 39.2 percent, also caused by the high operating temperature.

PEBBLE BED

A different type of high temperature reactor called the pebble bed reactor has been developed in Germany.[8,12] The AVR at Julich, a 15 MWe pebble bed reactor, has been operating since 1966 as a prototype for the 300 MWe thorium high temperature reactor (THTR) being built at Uentrop. In this type of gas cooled reactor the fuel elements are coated uranium-thorium carbide particles embedded in a six centimeter diameter spherical matrix of graphite. These spherical fuel elements, numbering about 75,000 in the AVR, are arranged in a loose statistically random heap within a surrounding core structure of graphite and carbonbrick. Refueling takes place under load, in that fuel elements are continuously being removed from the bottom of the pile. Unspent spheres and new spheres are added at the top and spent fuel is removed from the cycle.

The cooling gas is helium at 11 atm ranging from 270°C to 850°C producing steam at 505°C and 70 atm. Control is obtained by moving control rods in the reflector. However in the THTR, shutdown is obtained only by inserting control rods directly into the pebble bed. Emergency shutdown is by injecting boron trifloride gas into the core which at one atmosphere is sufficient to bring the reactor subcritical.

UHTREX

Since high temperature is one of the advantages of a gas cooled reactor, it is instructive to examine the results of the Ultra-High Temperature Reactor Experiment (UHTREX).[8,13,14,15] This experiment is a continuation of the Los Alamos TURRET reactor experiment which was originally proposed as a nitrogen cooled reactor to drive a closed cycle gas turbine. This proposal was revised in 1961 to a helium cooled reactor having an operating pressure of 34 atm, an exit temperature of 1320°C (2400°F), and no power conversion equipment.

The ceramic fuel particles are 93 percent enriched 147-208μ uranium carbide coated with a "Triplex" coating shown in Figure 2-2. The 20-30μ inner coating is a buffer layer of low density pyrolytic carbon to catch fission recoils, to dissipate stresses caused by differential dimensional changes, and to provide a void volume that will minimize fission gas pressure buildup. The 35-45μ middle layer consists of isotropic pyrolytic carbon with low permeability to fission gases and good resistance to radiation damage. The outer layer of approximately 40μ is fine grained columnar (granular) pyrocarbon. The interface between the isotropic and granular layers is intended to retard the propagation of any cracks that might develop.

These coated particles are embedded in a graphite matrix in the form of a one inch diameter, five and a half inch hollow cylinder. Four elements rest end to end in each of 312 radial fuel channels in the cylindrical graphite core. Refueling may be continuous under load by rotating the core to align the channel with the reloading line (as

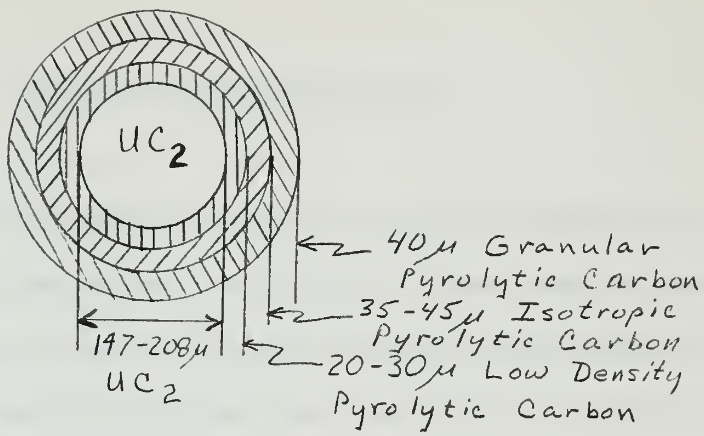

Figure 2-2

UHTREX Coated Particle

in a turret) and pushing in another fuel element. This forces the innermost fuel element to drop into the hollow center of the core where it falls out of the core into a discharge chute. A double valve arrangement is used at both the fuel element loading and discharge locations to prevent loss of coolant.

This reactor operated in 1969 at the design temperature of 1320°C. When used to heat steam in a Rankine cycle steam cycle, there is little advantage to be gained by operating above 800°C . However, outstanding efficiency would be achieved in the 900°C-1000°C range with a Brayton cycle gas turbine plant. The even higher temperatures obtained with this reactor could be used in process heat applications such as coal gasification.

III. EXISTING MARITIME REACTORS

It is instructive to examine the existing maritime reactors even though they are all of the pressurized water type. These reactors have given good service under the rough conditions on board ship as proven by the fact that over 200 such plants have been installed in military vessels, mostly in submarines but in some surface naval vessels also. Among the decisive advantages of nuclear power for naval vessels are the very great range of the ships and the fact (for submarines in particular) that the nuclear propulsion systems require no oxygen and emit no exhaust gases.

However, with merchant vessels, these advantages are much less important. The decisive consideration is economy in comparison with ships propelled by conventional means. In contrast to central station power plants, nuclear ship propulsion systems have not yet achieved an economic breakthrough. This is clearly illustrated by the fact that only a few ships have been built for peaceful uses and one of them, the N. S. SAVANNAH is already out of commission. Numerous studies have indicated that with larger ships, nuclear power probably provides an economic advantage.

N. S. LENIN

A peaceful use requiring the long range of the nuclear ship is the icebreaker. The N. S. LENIN, commissioned in 1959 by the Soviet Union, is powered by three pressurized water reactors, each capable of

producing 90 MWth, although two can provide the required 40,000 SHP.[16] The fuel elements consist of uranium dioxide clad with zirconium alloy. The fuel assembly contains a burnable poison in the form of boron-10 alloyed with the zirconium structural parts. These reactors produce steam at 28 atm and 307°C. This ship has apparently given good service and it is believed that two further nuclear powered icebreakers are now under construction in the Soviet Union.[17]

N. S. SAVANNAH

In 1962 the United States commissioned the N. S. SAVANNAH as a combined cargo and passenger ship. It was decommissioned in 1970 even though it was shown to be technologically reliable. It was not economically competitive, however, provoked numerous labor problems, and could not be viably operated without federal subsidy. The pressurized water reactor operated at 80 MWth to produce 22,000 SHP. The fuel was an average 4.4 percent enriched uranium dioxide clad with 304 stainless steel. The primary coolant operated at 123 atm between 257°C and 271°C which produced saturated steam at varying pressures and temperatures to the turbines depending on the load.[18,19]

N. S. OTTO HAHN

The N. S. OTTO HAHN is an 11,000 SHP ore carrier commissioned in West Germany in 1969. The 42 MWth reactor is based on the Babcock and Wilcox Consolidated Nuclear Steam Generator (CNSG-I), in that it includes self pressurization and an integrated structure, i.e., the primary heat exchangers are within the pressure vessel. The fuel is

an average 4 percent enriched uranium dioxide pellets clad with a chromium-nickel-niobium alloy. The primary coolant operates at 63 atm pressure between 266°C and 278°C producing superheated (36°C) steam at 31 atm and 273°C.[16,20,21]

N. S. MUTSU

The 10,000 SHP Japanese nuclear research ship N. S. MUTSU, which will probably be commissioned in 1973 also utilizes a reactor based on the CSNG-I. It is a 36 MWth pressurized water reactor. The fuel elements are an average 4 percent enriched uranium dioxide pellets clad in 304 stainless steel. The primary coolant operates at 110 atm between 271°C and 285°C producing steam at 38.5 atm and 246°C.[17]

N. S. ZAN THAN

It has been reported, but not confirmed, that the Peoples Republic of China has commissioned a 400 passenger, 23 knot nuclear ship, the N. S. ZAN THAN, powered by a 160 MWth pressurized water reactor.[22] No other information has been published to date.

IV. PROPOSED MARITIME GAS COOLED REACTORS

There have been several proposals specifically directed toward the use of gas cooled reactors for maritime use. Almost twenty years ago, the United States Maritime Administration instigated studies under the Maritime Gas Cooled Reactor (MGCR) Program. The culmination of the program was the prototype Experimental Beryllium Oxide Reactor (EBOR); however, funding was terminated in 1966 before the prototype was fueled. In the late sixties, Germany proposed to build a graphite moderated, helium cooled, direct cycle prototype at Geesthacht. This was intended to be primarily a central station prototype with the secondary objective being maritime development. Again, funding was halted before the project could get started. It is instructive to examine in detail some of these proposals.

GENERAL MOTORS DESIGN

In 1957 a design of a 20,000 SHP gas cooled reactor, closed cycle gas turbine system was produced by General Motors for installation in a 38,000 ton DWT tanker.[23] The 50 MWth reactor was to be graphite moderated and reflected and helium cooled with the coolant operating at a nominal 68 atm between 410°C and 705°C in the reactor. The fuel element operating temperature was the limiting factor of the design.

The 112 fuel elements were formed by two concentric rings (approximately 6.35 cm and 7 cm diameter) of clad plate with a graphite rod center. This was then inserted into a hole in the center of a

12.7 cm by 12.7 cm square graphite block. The helium path was in the gaps between the center rod, the plates, and the square block. The fuel element design may be seen in Figure 4-1. The fuel rings were to be a 93 percent enriched uranium dioxide dispersion in a .635 mm 316 stainless steel matrix clad with .25 mm of unfueled 316 stainless steel. The maximum fuel element surface temperature was 870°C. Control was by 21 europium oxide in stainless steel cruciform shaped rods, gravity inserted. The cruciform webs were 12.7 cm by 12.7 cm placed as indicated in Figure 4-2.

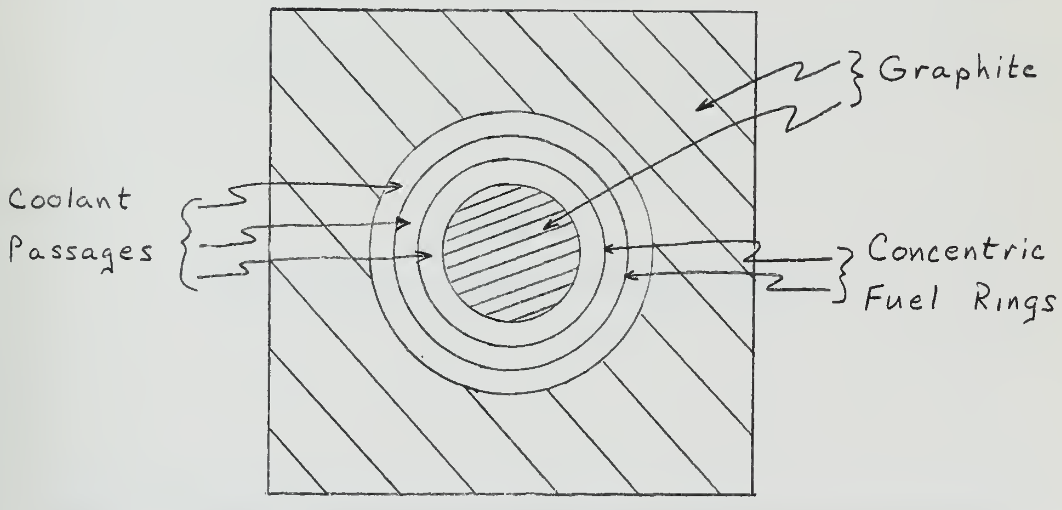

Figure 4-1

Fuel Element for General Motors Reactor

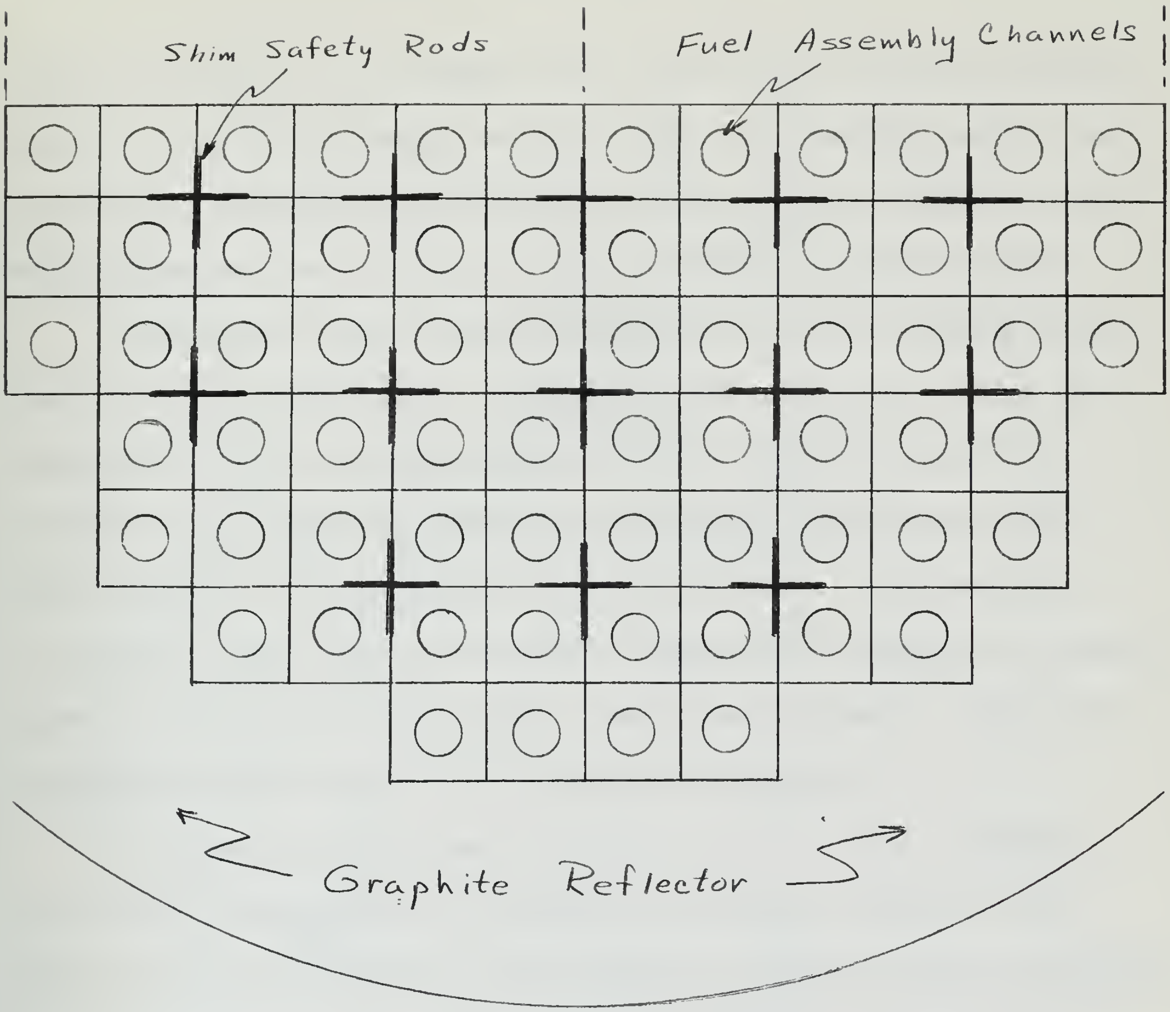

Figure 4-2

Core Layout for General Motors Reactor

The cool (410°C) helium entered the reactor pressure vessel at the bottom and cooled the pressure vessel by flowing upward between it and the graphite reflector. No thermal shields were used. The helium then entered the top plenum chamber and flowed down through the individual fuel elements and emerged from the bottom at 705°C. The

active core was 1.5 m in diameter by 2 m high. The 9 cm thick pressure vessel was 2.5 m in diameter and 8.7 m high. The overall reactor was to be 6 m in diameter by 11.3 m high including the top shield of lead and polyethylene and the side and bottom shields of water-iron-lead.

A particular safety problem anticipated was the flooding of the core in case of a collision. Assuming maximum water in the core (no steam voids), it was calculated that positive reactivity ($\Delta k/k$) of +0.09 would be added because of filling the gas cooling spaces with a better moderator. This calculation was made with a void fraction of 8.7 percent. This positive reactivity addition was assumed to be easily taken care of by the control rods. Some recent designers, particularly in Germany, consider this to be a severe safety hazard.

A schematic of the power conversion cycle is shown in Figure 4-3 and the corresponding T-s (temperature-entropy) diagram of the cycle is shown in Figure 4-4. The numbers on the T-s diagram correspond to the gas conditions at the corresponding physical locations in the schematic. A closed gas turbine cycle has only four essential components - a compressor, a heat source, a turbine, and a heat sink. An engine based on this minimum number of components has the advantage of simplicity but the decided disadvantage of poor thermal efficiency. To improve performance the cycle must become more complex. (See Appendix A for a more complete review of thermodynamic cycles). As can be seen in the schematic, the General Motors design compromised by including a regenerator and three separate compressors with intercooling between them. The study reports an overall efficiency of 31 percent with the

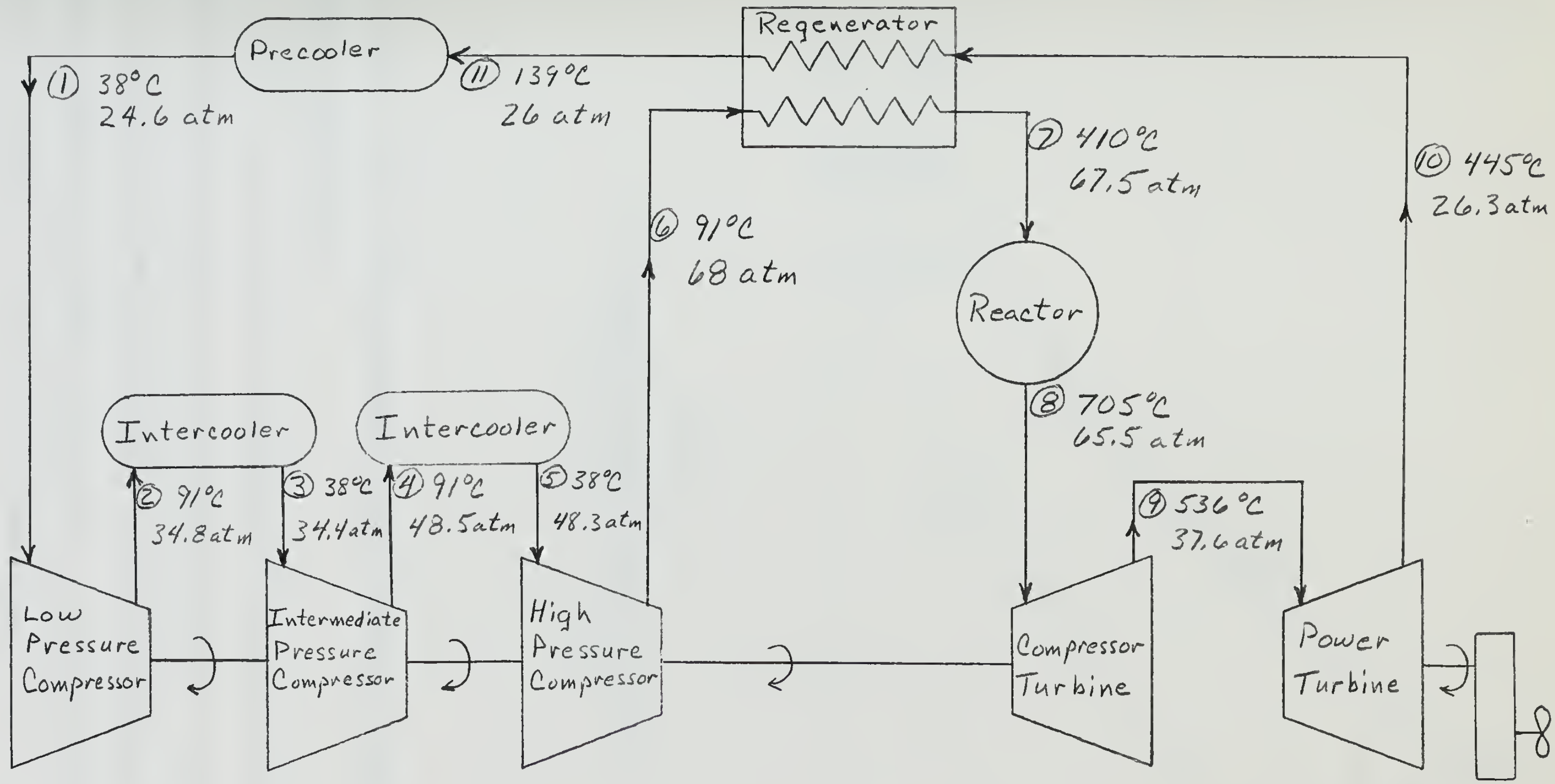

Figure 4-3

Schematic of Power Conversion Cycle for General Motors Design

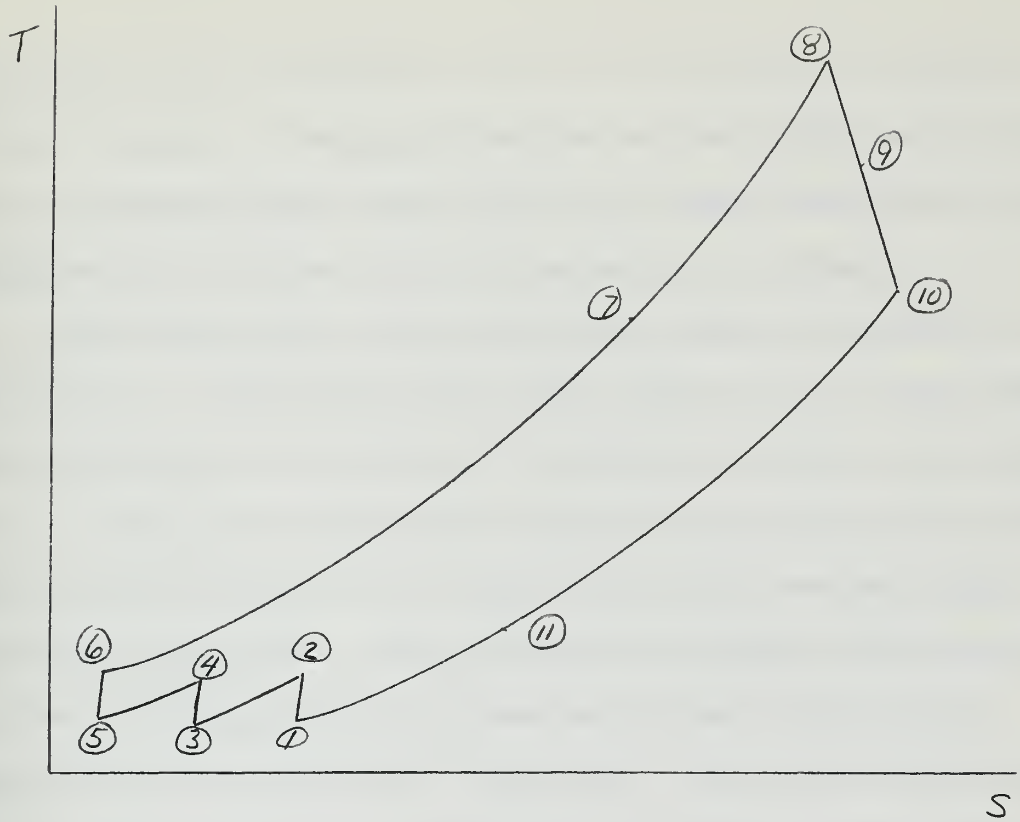

Figure 4-4

T-s Diagram of Power Conversion Cycle
for General Motors Design

then present (1957) technology with over 40 percent possible with increased temperatures.

Helium was chosen over nitrogen and carbon dioxide as the working fluid for reasons of its good high pressure thermodynamic properties, good heat transfer properties, good turbomachinery properties, and good chemical and nuclear properties. The study indicates that, as of 1957, the behavior of helium in turbomachinery was an unknown, but that few problems were anticipated. It should be pointed out that

large scale helium turbomachinery has yet to be built, but the technology is growing. A design problem that was anticipated in 1957, but has since apparently been solved, was that of shaft lubrication and sealing while preventing coolant contamination and leakage.

The proposed power conversion control system was divided into two parts: 1) a "power available" control which set the reactor power level and therefore the compressor and compressor turbine power level, and 2) a "power delivered" control which set the portion of the available power actually used at the power turbine, the remainder being dissipated in the precooler. The power available depends upon the rate of gas flow, which depends upon the compressor speed, which in turn depends upon the reactor outlet temperature. Since rapid reactor power level changes could produce undesirable side effects, it was anticipated that as much as an hour would be required to bring the engine from idle to full load, if the reactor was at idle power. However, in extreme emergency, the reactor could be changed from idle to full power in a matter of minutes. Once the reactor level was at full load, the engine availability would be almost instantaneous.

The power actually delivered to the shaft could be decreased from the level available by a combination of bypassing a portion of the working fluid around the power turbine and by inserting additional resistance in series with the power turbine in the form of a throttle valve. Astern power was produced by reversing the pitch of the propeller since any other method is difficult using a unidirectional gas turbine.

MGCR/EBOR

In 1958, General Atomic Division of General Dynamics Corporation (now Gulf General Atomic) was awarded a contract to develop the technology required to evaluate a gas cooled reactor for maritime use.[24,25] The program as initiated was called the Maritime Gas Cooled Reactor (MGCR) Project. The turbomachinery development was subcontracted to Westinghouse Electric Corporation.[26] The MGCR program was terminated in 1963 but reactor development was continued under the Experimental Beryllium Oxide Reactor (EBOR) Program. Subsequently, this program also was terminated in 1967 before fuel was inserted into the prototype.[27] The results of the program are presented in chronological order.

In 1958, as a result of comparison studies between reactors that were: a) graphite moderated, helium cooled, b) graphite moderated, carbon dioxide cooled, c) water moderated, carbon dioxide cooled, and d) zirconium hydride moderated, carbon dioxide cooled, the graphite moderated, helium cooled reactor was selected for further development. Fueling materials considered were both diluted and undiluted metal clad ceramic type fuels such as uranium dioxide, uranium dioxide diluted with aluminum or beryllium oxide, and uranium carbide diluted with graphite.

In 1959, it became apparent that the neutron leakage from the relatively small graphite moderated reactor being considered was too high for it to compete economically with other proposed nuclear propulsion systems. This fact, coupled with information that the cost of

beryllium oxide shapes could be reduced, provided the shapes were suitably simplified, indicated that the choice of moderator should be reconsidered, and a helium cooled beryllium oxide moderated reactor was selected after more comparative evaluations. An important aspect of using beryllia as a moderator was the fact that by elimination of graphite in the reactor, the carbon mass transport reaction with impurities in the coolant was no longer a problem. The emergency cooling problem was also simplified since air could be introduced into the reactor immediately after shutdown without detrimental oxidation reactions. It also became possible to consider water as a coolant in the ultimate emergency since beryllium oxide and water undergo no appreciable reaction at the designed operating temperatures. However, the reactivity change caused by flooding this core was not discussed in the literature.

The turbomachinery development program by Westinghouse was originally based on a 20,000 SHP output with maximum cycle conditions of 705°C and 55 atm. Subsequently, the pressure was increased to 78 atm with possible temperature operation at 816°C included in the design. Nominal power output was increased to 30,000 SHP. At the original design point, a cycle similar to the General Motors design was selected but only two compressors and one intercooler were used. The power turbine was operated at lower pressure and temperature than the compressor turbine which directly followed the reactor in the cycle.

At the higher temperatures and pressure, it was found more practical to derive output power from the high pressure turbine, using the low pressure turbine to drive the compressors. This system is

referred to as the "high pressure drive" scheme as contrasted to the "low pressure drive" originally used. The difference between the two systems becomes more significant as cycle pressure level and temperature increase. This results from the desirability of operating the compressor turbine at high speed to minimize the number of compressor stages required. Since stress level varies as the wheel speed squared, the compressor turbine will operate at much higher stress levels than the power turbine which is typically operated at a much lower speed. Therefore, to optimize stress distribution, the high pressure drive scheme was used.

A schematic of the power conversion system is shown in Figure 4-5 with the corresponding T-s diagram shown in Figure 4-6. The numbers relate the T-s diagram to the physical plant schematic. The cycle parameters are: power output - 32,869 HP, maximum temperature 816°C, maximum pressure - 78 atm, and cycle efficiency - 37.06 percent. Note, that there is a significant improvement over the General Motors design. More improvement at higher operating temperatures is also possible.

The final reactor to be used in this program was the beryllium oxide moderated and reflected, helium cooled type as typified by the EBOR prototype. The fuel elements were never inserted in the prototype because of cancellation of the program. However, they were designed to be made of 62 percent enriched uranium dioxide pellets in a beryllium oxide matrix clad with Hastelloy X, a low cobalt, nickel base alloy. The assembly consisted of an annular ring of eighteen rods around a

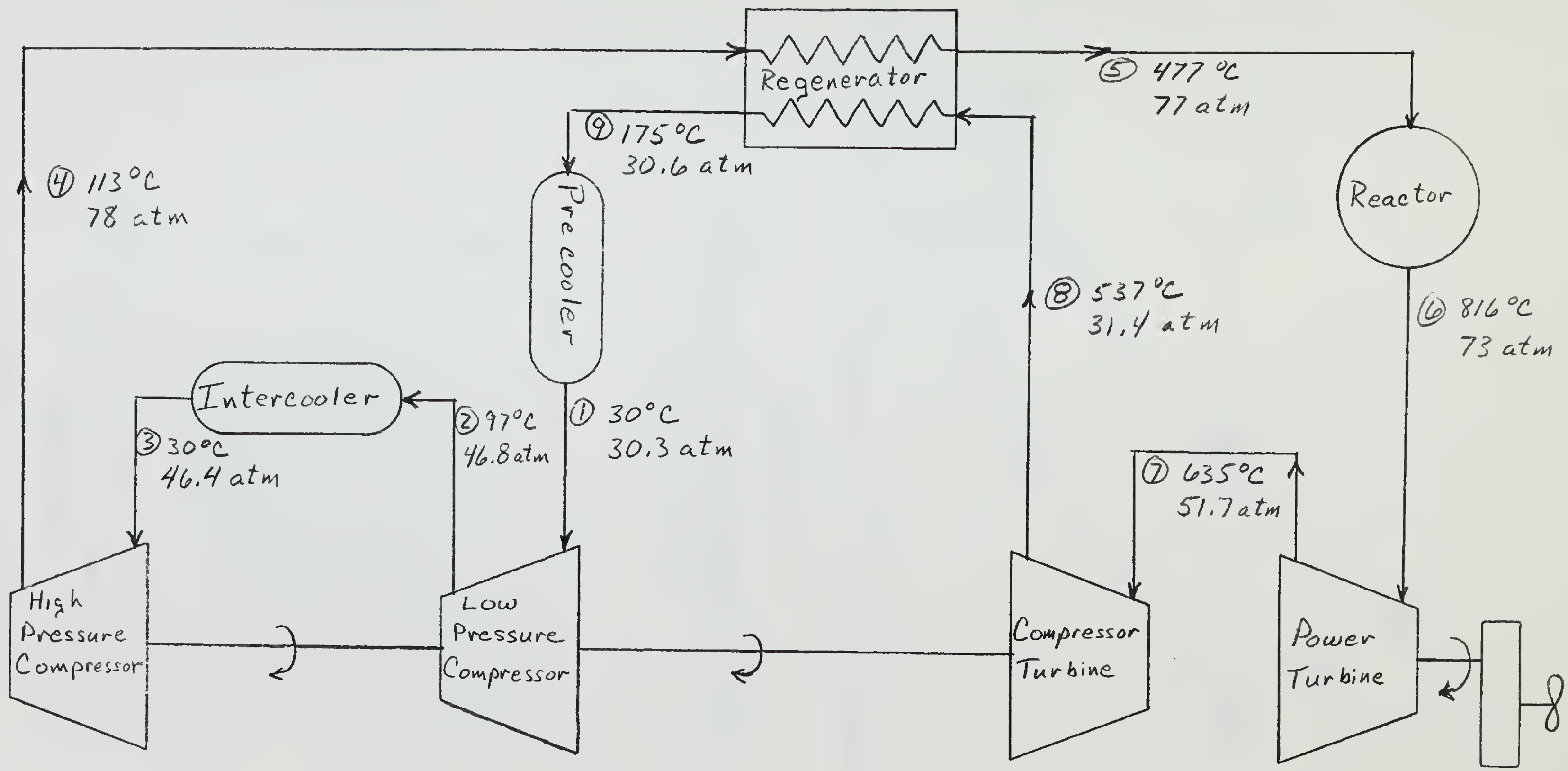

Figure 4-5

Schematic of MGCR Power Conversion Cycle

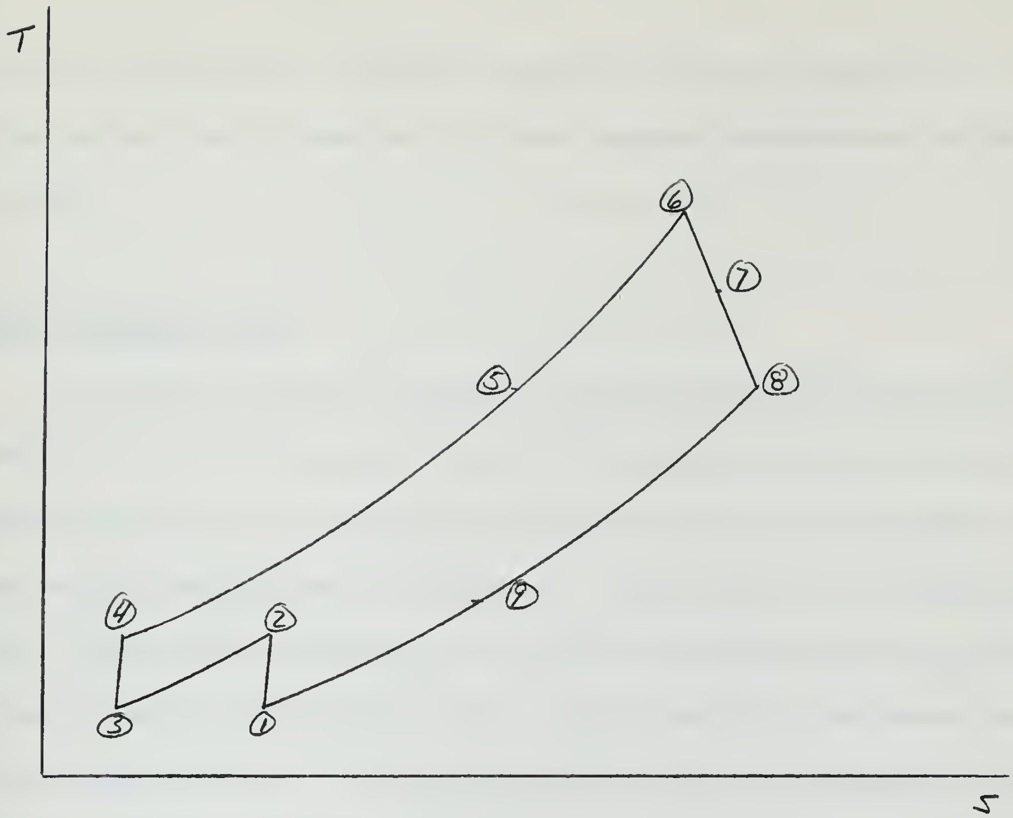

Figure 4-6

T-s Diagram of Power Conversion Cycle
for MGCR Design

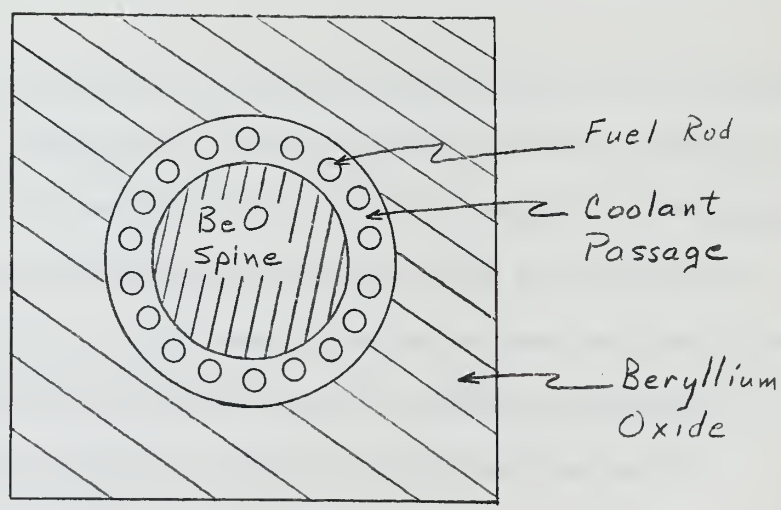

Figure 4-7

EBOR Fuel Assembly

beryllium oxide spine. The fuel assembly is shown in Figure 4-7. The helium coolant was to operate at 75 atm between temperatures of 400°C and 700°C.

GENERAL ELECTRIC 630A

In 1961, the General Electric Corporation began design work on a maritime gas cooled reactor based on their work with the Aircraft Nuclear Propulsion plant. This work resulted in a reactor called, "The 630A Maritime Nuclear Steam Generator." The 630A was a uranium dioxide fueled, light water moderated, air cooled, beryllium oxide and graphite reflected reactor rated at 60.4 MWth, which produced superheated steam in an integrated boiler. It underwent five major modifications and by 1965 the 630A Mark V emerged in two forms, "A" and "B". The Mark V (A) was an air cooled, water moderated, calandria type reactor while the Mark V (B) was a helium cooled, water moderated, tube type reactor. Since the designers favored the Mark V (B), it is the one presented in detail.[28]

The fuel rods were uranium dioxide pellets enriched to 6 percent and clad in Incoloy. There were 27 rods per fuel assembly, arranged in a 1.1 cm triangular pitch and 216 fuel assemblies arranged in hexagonal patterns around 91 moderator tubes. The active core was hexagonal in shape, .66 m on a side and 1.07 m long. The active core was surrounded by an inner beryllium oxide reflector, with an average thickness of 12 cm, followed by a 20 cm graphite reflector, which was pierced by cooling holes and separated from the inner reflector by a 2.5 cm annulus. The voids were filled with helium coolant.

A cross sectional view of a section of the 630A Mark V (B) core can be seen in Figure 4-8. The coolant flow was from the circulator up the outside of the core, through the annulus and the graphite reflector, into the coolant inlet flow plenum. From there, down through the active core through the boiler and back to the circulator.

The reactor was controlled by means of concentric Incoloy shim rods which fit into the moderator tubes to displace the water for their reactivity effect. The control by water displacement resulted in an increase in the mean energy of the neutron spectrum with the resultant neutron absorption by the U-238. Thus, the excess reactivity was held in the conversion of fertile to fissile fuel rather than in a burnable poison. Also, because each moderator tube could be shimmed separately, a method of gross radial power flattening was provided. In addition a single concentric sheath of borated stainless steel was included in 48 of the moderator tubes as safety rods and inserted for positive shutdown during a scram. This type of moderator tube is shown in Figure 4-8 with all the shim rods inserted.

A shield plug assembly consisting of the active core, beryllium oxide reflector, shield plug, and control mechanism, as a single unit could be removed from the reactor intact. This arrangement would facilitate refueling and/or repair.

Power conversion was by conventional steam turbomachinery. The primary coolant, helium, operated from 290°C to 650°C at 56.5 atm producing steam at 102 atm and 538°C. The overall conversion efficiency was calculated to be 33.7 percent.

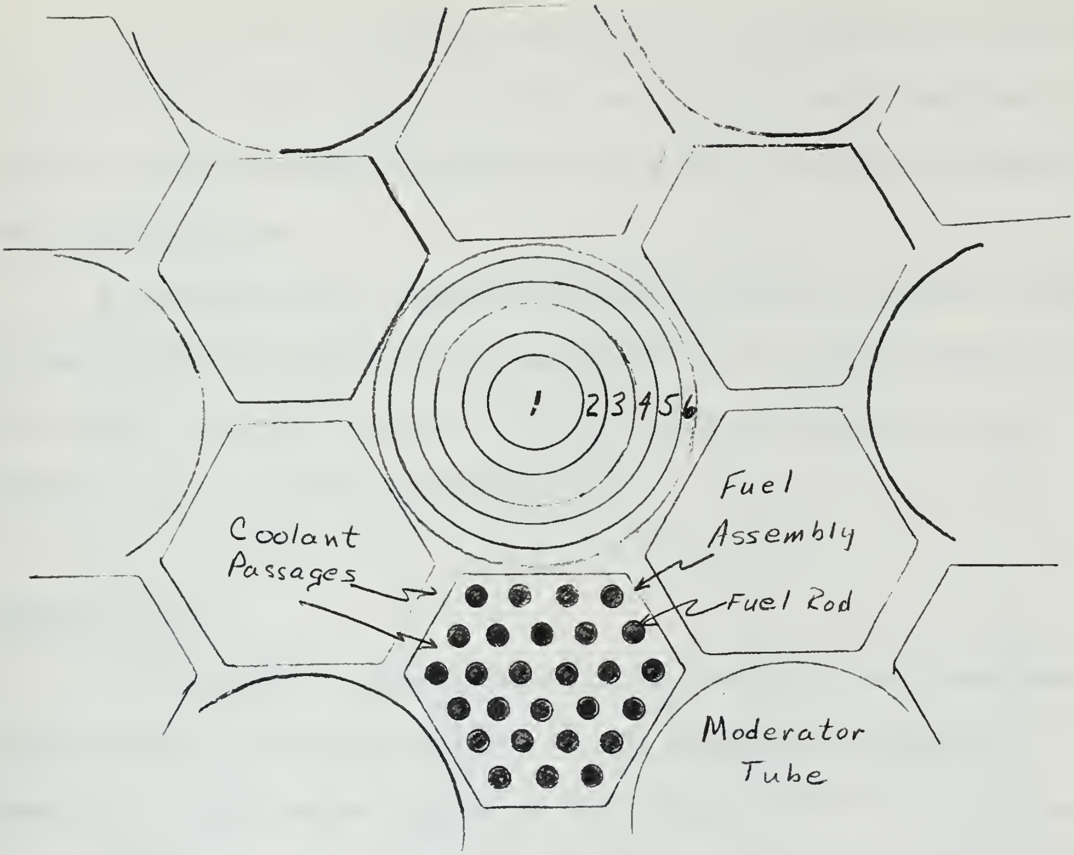

Figure 4-8

Core Layout of 630A Mark V (B)
Moderator and Fuel Assembly

1. Inner water displacement shim
2. Outer water displacement shim
3. Shim tube
4. Safety tube
5. Water
6. Moderator case

A preliminary calculation was made on the problem of flooding the core and it was found that a positive reactivity of +0.10 $\Delta k/k$ would result which the designers felt was well within the control range of the designed system.

A design has been suggested utilizing this basic reactor design but employing a closed cycle gas turbine.[29] A simplified schematic of that system is shown in Figure 4-9. The numbers correspond to the numbers on the T-s diagram of Figure 4-6.

GEESTHACHT KSH

In 1968, GHH, a consortium of German industrial and government organizations, proposed to build a 24 MWe direct cycle, graphite moderated, helium cooled reactor at Geesthacht.[30,31,32,33] This was to be a prototype of both a central station plant and a maritime propulsion system. However, the construction contract was never issued partially because of the anticipated safety hazard at sea of flooding the core.[34] The design is presented here for completeness.

The reactor core consisted of 657 cylindrical graphite fuel elements similar to those of the Peach Bottom prototype without the purge channels. Each element was made up of a graphite sleeve enclosing a graphite matrix containing fissile and fertile coated particles similar to those used in the Fort St. Vrain reactor. The active core was surrounded by a graphite reflector. The coolant pressure was 25 atm and coolant temperatures in the reactor ranged from 425°C to 735°C.

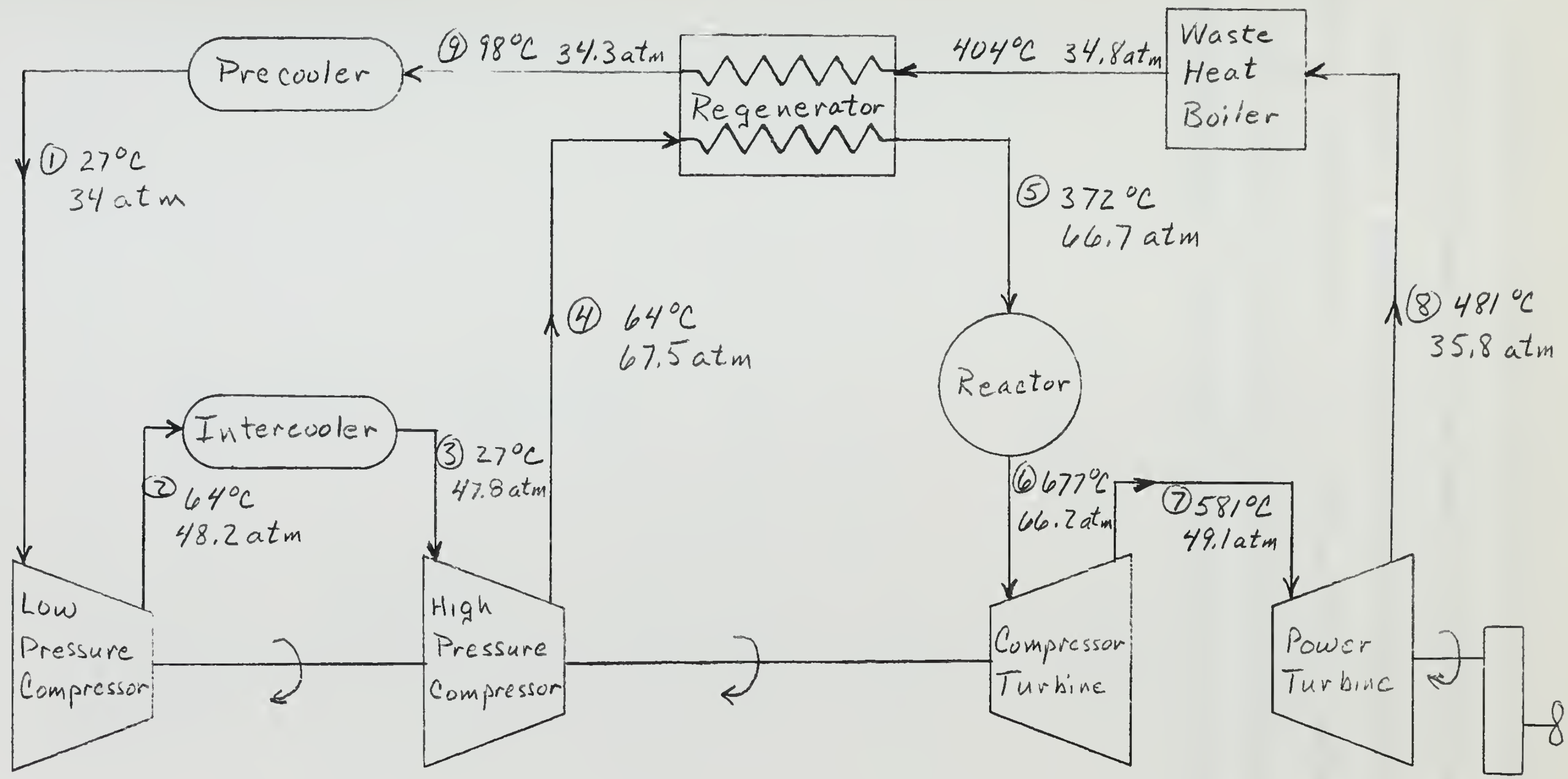

Figure 4-9

Proposed Power Conversion Schematic for 630A

A schematic of the power conversion system is shown in Figure 4-10. Detailed information on the design particulars was not obtainable. Note that a single turbine is used and the generator and compressors are all connected to it -- the generator through a reduction gear. Note also that the design contains two intercoolers and three separate compressors. The overall efficiency of the design is 37 percent.

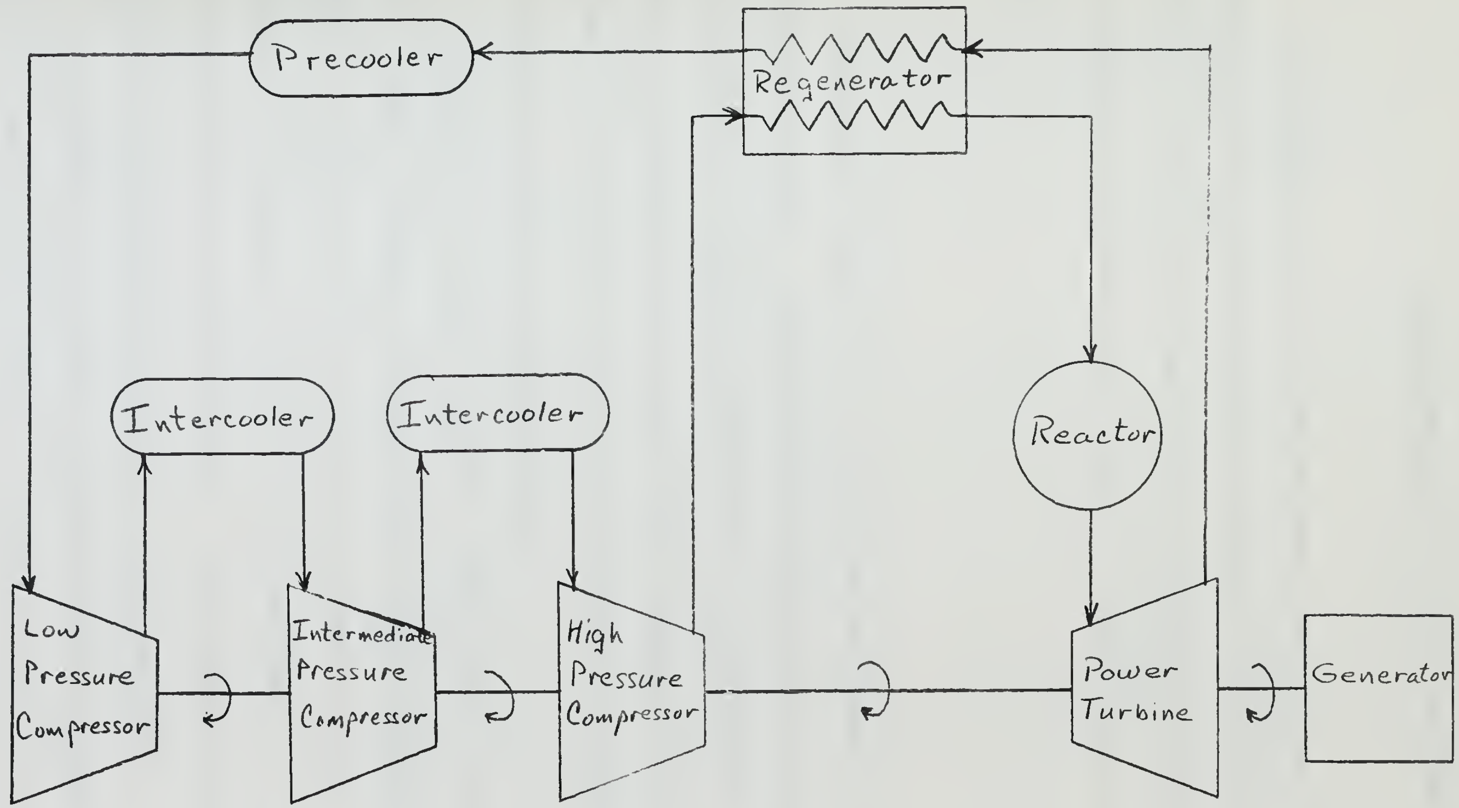

Figure 4-10

Power Conversion Schematic of the Geesthacht KSH Design

V. RELATED MOBILE GAS COOLED REACTORS

The present status of central power station gas cooled reactors and existing and proposed maritime propulsion systems has been reviewed. It is also instructive to examine the technology developed in some programs related to maritime use in that the systems were intended to be mobile.

ML-1

The United States Army, through Aerojet General Nucleonics, attempted to produce a mobile nuclear power plant on skids.[35,36] The Gas Cooled Reactor Experiment (GCRE) was a heterogeneous, water moderated, nitrogen cooled reactor with a nominal output of 2 MWth. It operated from early 1960 until April 1961 when the pressure vessel failed and the reactor was deactivated.

The Mobile Low-power Plant No. 1 (ML-1) was then developed. It was a 3.3 MWth, .33 MWe, nitrogen cooled, water moderated reactor utilizing a closed cycle gas turbine for power conversion. The fuel elements were 93 percent enriched uranium dioxide in a matrix of beryllium oxide clad with Hastelloy X. The nitrogen operated in the reactor at a nominal 20.5 atm between 422°C and 650°C.

A schematic of the power conversion cycle is shown in Figure 5-1. Note that a single turbine is used and that no intercooling is included. Consequently the overall efficiency of the cycle is a low ten percent.

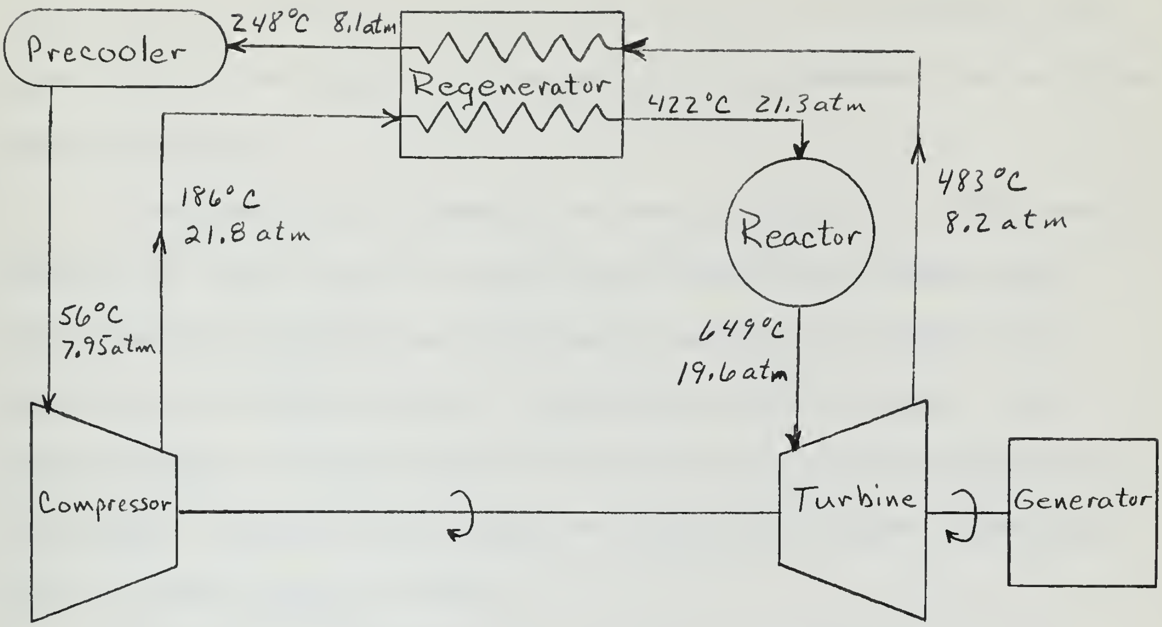

Figure 5-1

Power Conversion Schematic of the ML-1

This program was relatively successful in that a prototype was built and operated for three years. However in 1966, before complete design objectives were met, funding was terminated because of lack of a specific mission.

ANP

A program that developed many ideas although it was cancelled in 1961 was the Aircraft Nuclear Propulsion (ANP) Program.[37] At the present time, there is a low level effort by NASA to solve some of the problems associated with nuclear aircraft. The key problem is safety -- crashing an operating reactor into the ground at a high velocity. The

results of this program are directly applicable to the maritime field, especially the low weight to horsepower ratio requirements and collision safety solutions.

A proposed system schematic is shown in Figure 5-2. The reactor consists of a gas cooled (or liquid metal cooled) core, high burnup fuel, complete shielding and containment, and a heat exchanger that heats the air flowing through a conventional turbofan engine. This heat exchanger is placed directly in front of the normal combustors for the engine. Therefore, the engine can then run on either conventional fuel or nuclear energy, or both.

Some of the key developments in this program are the impact survival designs for the containment. These have been considered for impacting instrumental payloads on the moon and planets in the space program. The three most promising methods for absorbing kinetic energy are to utilize the crushing of balsa wood, the deformation of frangible tubes, and the crushing of honeycombs of metal or plastic. It is proposed that this technology could be applied to collision of sea vessels, also, in order to reduce the possibility of rupturing the reactor containment.

Another interesting development is the concept of high burnup fuel to provide for long life reactors in order to reduce core inventory and core size. The concept of a very high burnup (20 percent) vapor-transport fuel pin is being tested. The pin consists of a tube that is designed as a pressure vessel. Fuel in the form of uranium dioxide is contained within the pin in a thin layer relative to the thickness of the tubular pressure vessel. The objective is to assume

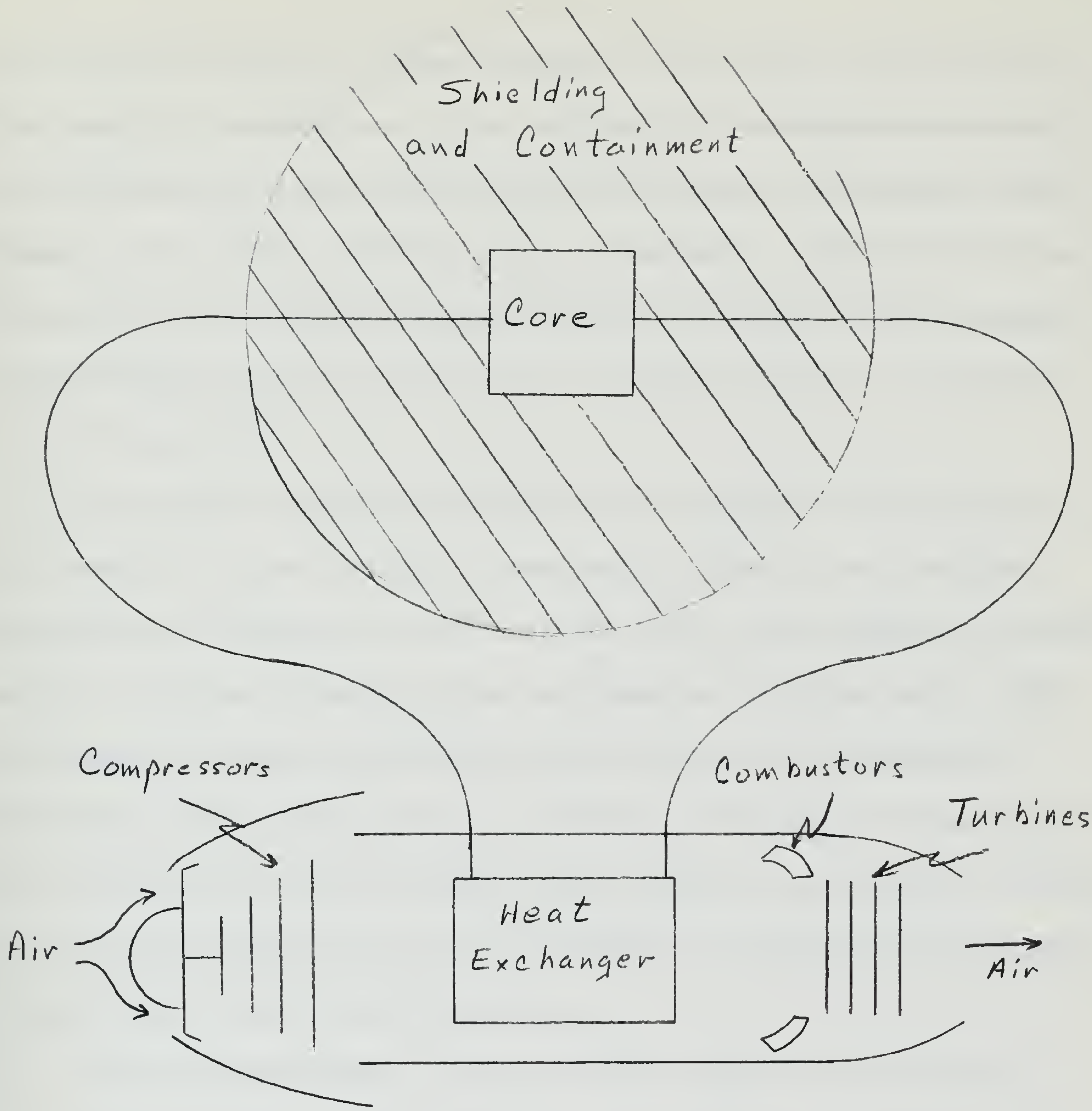

Figure 5-2

A Recent Proposal for a Nuclear Aircraft Power Plant

that the fuel material is weak compared to the clad so that when the fuel swells or expands due to the buildup of fission products within it, the fuel will flow plasticly into the central void without introducing a major stress in the strong clad material. The void also provides room for the gaseous fission products to expand and is designed large enough to hold the fission gas at desired burnup at a pressure the clad can stand.

A property of these fuel pins may be applied to the control of the reactor. If the fuel pin is designed to operate such that the temperature of the inside surface of the fuel is above 2200°C, the fuel tends to vaporize in hotter areas and condense in cooler areas. This would tend to eliminate hot spots, balance the overall change in reactivity of the reactor over its lifetime, and provide unique ways of controlling fuel distribution over the lifetime of the reactor. It may be pointed out that the Fort St. Vrain HTGR is designed for 20 percent burnup using ceramic coated particles.

A recent unpublished proposal by Westinghouse utilizes this technology as a basis for developing the required low weight to horsepower nuclear propulsion systems needed for advanced surface effect ships.[38] The information contained in Table 6-4 is from this proposal.

NERVA SOLID CORE ROCKET

Another program which has yielded much new technology is the nuclear rocket program (NERVA).[39,40] The uranium carbide, ceramic clad fuel particle was originated in this program and has been previously discussed in relation with the high temperature gas reactors. A NERVA

Technology Reactor (NTR) has been proposed to provide power for a space station.[41] This space reactor is a helium-xenon cooled 2.2 MWth reactor based on the hydrogen cooled 1100 MWth NERVA experimental engine which was very successful. Since the NTR design is more applicable to maritime use it will be the design discussed. A simplified diagram is shown in Figure 5-3.

The core of the NTR consists of 93 percent enriched uranium carbide, ceramic clad fuel particles in a matrix of graphite. The 673 fuel elements are hexagons 1.9 cm across the flats and 52 cm long. Each element has seven holes for coolant flow. The elements are packed together to give an equivalent core diameter of 52 cm. Around the periphery is a graphite barrel and then a beryllium reflector. The reflector structure is composed of eight beryllium segments, each of which contains a rotatable control drum. Each drum contains a poison plate of boron-copper alloy. The drums are rotated by actuators similar to an electric stepper motor. The extreme compactness of this reactor can be seen from its design weight of 6000 Kg (excluding shield) and design size of 1 m diameter by 1.6 m length. The coolant operates at 15.5 atm between temperatures of 627°C and 872°C. The coolant moves up through holes in the reflector and down through holes in the core.

The proposed power conversion system is a small 15 KWe Brayton cycle gas turbine developed by the NASA Lewis Research Center. These small converters would be ganged in parallel to produce the needed total power. It is expected that the turbomachinery development will lead to a capability of a 1150°C turbine inlet temperature. A mixture of gases is used as the working fluid because high molecular weights

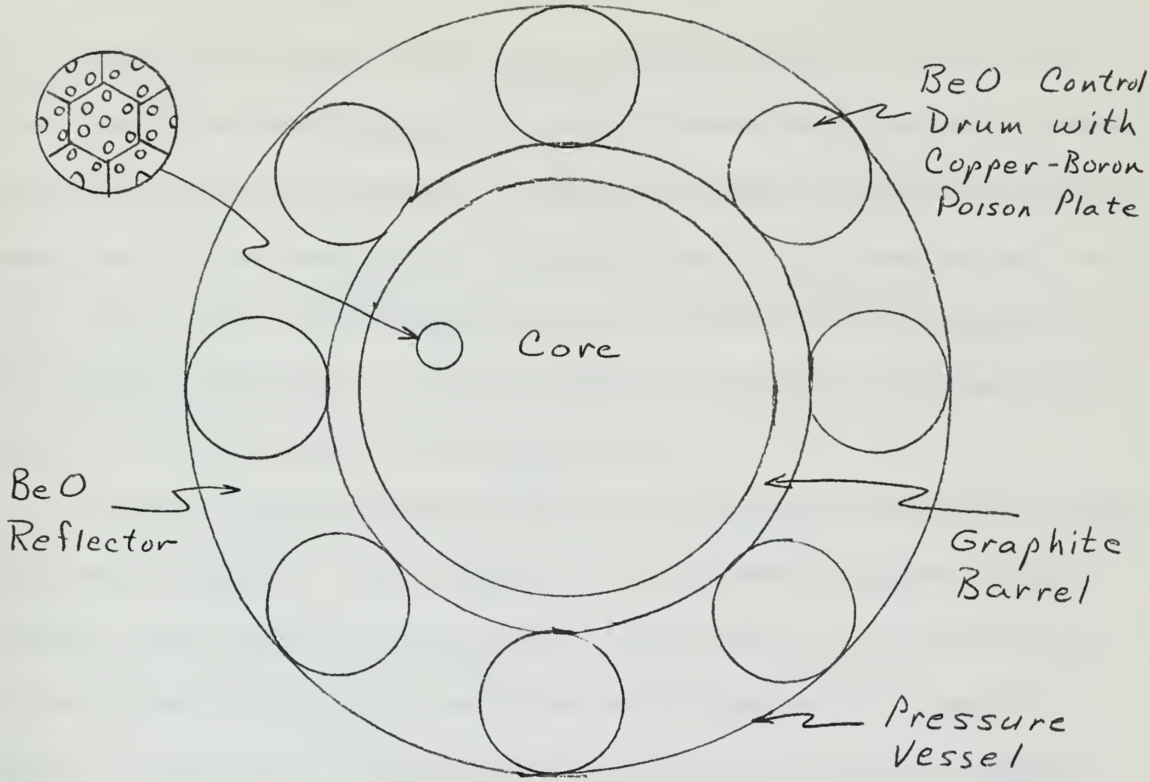

Figure 5-3

Simplified Diagram of the NERVA Technology Reactor

are desired for small turbine size, but low molecular weights are needed for good heat transfer. A mixture of helium and xenon with a molecular weight of 39.94 is used here as a compromise.

PLASMA CORE REACTORS

A different approach in the nuclear rocket field is to replace the solid core of fuel with a gaseous core.[42,43] A particular problem

in evaluating the plasma core reactor concepts derives from the fact that a small-scale prototype test reactor cannot easily be built because of critical mass requirements. A full-scale engine would produce power equivalent to the consumption of large cities, in a volume on the order of a cubic meter, with temperatures up to 100,000°C and a pressure of up to 1000 atmospheres. For this reason the majority of plasma core reactors are not feasible for shipboard use.

However a novel approach similar to an internal combustion engine has been proposed.[43] Since fission is an exothermic process, thermodynamic cycles such as the Otto cycle (See Appendix A) are envisioned. As indicated, steady state configurations of gaseous core reactors require large reactor dimensions and large critical masses. However, if enriched uranium fluoride (UF_6) gas is compressed in a transient fashion and if expansion of the fission heated gas is employed for heat removal and power generation, reactor size and critical mass can be greatly reduced.

A turbojet scheme is possible in which gaseous enriched UF_6 is passed through a moderator-reflector region, where it is compressed and becomes critical, and then is expanded to drive a turbine. After this the gas is recirculated to the reactor station.

A nuclear piston engine has also been proposed as shown in Figure 5-4. The engine has an intake stroke for drawing enriched UF_6 into a cylinder which is surrounded by a moderator-reflector. An exhaust stroke ejects the UF_6 plus fission products. Chain reaction is initiated by the neutron flux from an auxiliary source. An auxiliary precompression piston follows the working piston at high compression to

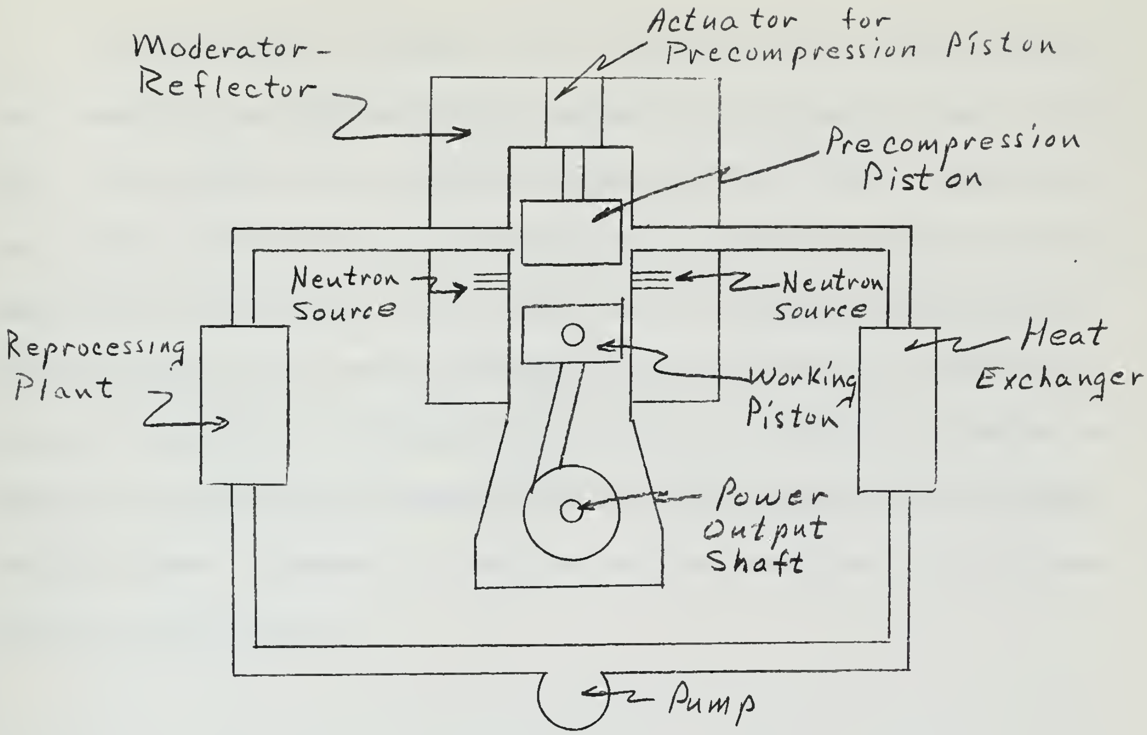

Figure 5-4

A Nuclear Piston Engine

provide time for the chain reaction to build up and to assure that maximum power is released as the working piston passes top dead center. Thereafter the reactor must be shut down rapidly to avoid release of fission heat after the working piston is already well into the next stroke. This is accomplished by retracting the precompression piston abruptly, causing a high neutron leakage. External equipment removes fission products, cools the gas, and recycles it back to the engine.

Thermodynamic efficiency can be improved with a UF_6-HE mixture as a combined fuel-working fluid, because of increased average specific heat. The cylinder has a minimum volume of 0.24 m^3 and a compression ratio of 10. The speed of the crankshaft is 100 rpm, and the critical mass is calculated to be 2.1-2.9 Kg U-235. Performance calculations indicate power up to several megawatts per cylinder at an efficiency of up to 60 percent. Problems include finding materials to handle the highly corrosive UF_6 gas and methods to rapidly initiate and stop the neutron fluxes. However, if the idea is ever developed it could have maritime applications.

VI. COMPARISON OF DISCUSSED SYSTEMS

The following tables give detailed data on each of the reactors discussed in this paper. Some of the data was not obtainable by the author and this will be indicated by an asterisk (*) and non-applicability will be indicated by a dash (-). The principle source of the information will be indicated by the superscript near the name.

Table 6-1

Technical Data on Existing Central Power Station Gas Cooled Reactors

Reactor	Magnox[44] Calder Hall	Magnox[44] Wylfa	AGR[44] Windscale	AGR[44] Hunterston B
Year Activated	1956	1971	1963	1973
Power (MWt)	268	3751	126.8	2988
Power (MWe)	50.5	1180	31.9	1243
Active Core Dia (m)	9.4	17.4	4.57	8.3
Active Core Length (m)	6.4	9.1	4.27	9.1
Avg Power Density (KW/l)	.55	.9	1.6	2.76
Power Conversion at (atm/°C)	Steam at 15/321	Steam at 45.3/401	Steam at 46.3/465	Steam at 158/538
Max Surface Temp (°C)	495	451	870	825
Max Fuel Temp (°C)	600	569	1700	1500
Coolant	CO_2	CO_2	CO_2	CO_2
Coolant Mass Flow (Kg/sec)	333	2560	119	470
Coolant Inlet Temp (°C)	150	247	260	318
Coolant Outlet Temp (°C)	345	414	460	648.5
Coolant Pressure (Atm)	7.7	27.2	19.4	41
Moderator	Graphite	Graphite	Graphite	Graphite
Fuel	U	U	UO_2	UO_2
Fuel Form	Rods	Rods	pellets	pellets
Fuel Enrichment (%)	Natural	Natural	3	2.4
Cladding	Magnox	Magnox	20/25 Nb ss	SS
Cladding Thickness (mm)	2.1	*	.25	.38
Control Material	Boron Steel	Boron Steel	Boron Steel	Boron Steel
Efficiency (%)	18.8	31.4	24.3	41.6

Table 6-1 (Continued)

HTGR[44] Peach Bottom	HTGR[44] Dragon	HTGR[44] Ft. St. Vrain	Pebble Bed[44] AVR	Pebble Bed[44] THTR	UHTREX[45]
1967	1964	1973	1967	1976	1969
115	20	842	51	750	3
40	-	330	13.6	300	-
2.77	1.07	6	3	5.6	1.78
2.29	1.6	4.6	3	6	1.0
8.3	14	6.3	2.5	6	1.3
Steam at 99/538	-	Steam at 164/538	Steam at 69.5/505	Steam at 175/530	-
1046	950	-	1050	1050	1593
1331	1500	1260	1250	1250	1610
He	He	He	He	He	He
60.5	*	107	9	49	1.3
344	350	400	270	262	870
728	750	775	850	750	1320
22.8	19.4	47.5	10.6	39.8	34
graphite	graphite	graphite	graphite	graphite	graphite
UC_2/ThC_2	various	UC_2/ThC_2	UC_2/ThC_2	UC_2/ThC_2	UC_2
coated part.	coated part.	coated part.	coated part.	coated part.	coated part.
93.2	93	93	93	93	93.5
Pyro-C	Pyro-C	Pyro-C	Pyro-C	Pyro-C	Pyro-C
-	-	-	-	-	-
B_4C	B_4C	B_4C	B_4C/Ti_4C	B_4C	*
35	-	39.2	28.3	40	-

Table 6-2

Technical Data of Existing Maritime Reactors
(All Pressurized Water Type)

Ship	Lenin[44]	Savannah[44]	Otto Hahn[44]	Mutsu[44]
Year Activated	1959	1962	1968	1973
Power (MWt)	3 at 90 ea	80	42	36
Power (SHP)	40,000	20,000	11,000	10,000
Active Core Dia (m)	1.0	1.6	1.15	1.15
Active Core Len (m)	16	1.7	1.12	1.04
Avg Power Density (KW/1)	*	23	33	33.5
Power Conversion at (atm/°C)	steam at 28/307	steam at various	steam at 30/273	steam at 37.3/246
Max Surface Temp (°C)	*	335	475	322
Max Fuel Temp (°C)	*	2350	1475	1770
Coolant	water	water	water	water
Coolant mass flow (Kg/sec)	*	*	222	*
Coolant inlet temp (°C)	261	257.2	267	271
Coolant outlet temp (°C)	317	271	278	285
Coolant pressure (atm)	*	119	61.5	106
Moderator	water	water	water	water
Fuel	UO_2	UO_2	UO_2	UO_2
Fuel Form	*	pellets	pellets	pellets
Fuel enrichment (%)	5	4.4	4.03	3.8
Cladding	Zr Alloy	304 ss	Cr Ni Nb	304 ss
Cladding thickness (mm)	.8	.89	.35	.4
Control Material	*	Boron Steel	B_4C	Ag In Cd
Efficiency (%)	*	20.5	19.5	20.8

Table 6-3

Technical Data of Proposed Gas Cooled Maritime Reactors

Reactor	General Motors[23] Proposal	General Atomic[45] EBOR	General Electric[24] 630A	GGH[32] Geeshtacht
Approx Year	1957	1962	1965	1967
Power (MWt)	55	10	60.4	65
Power (SHP)	20,000	-	27,300	24 MWe
Active Core Dia (m)	1.5	1.15	1.3	2.48
Active Core Len (m)	2	1.93	1.1	2.1
Avg Power Density (KW/l)	*	13.7	*	6.4
Power Conversion	closed cycle	-	steam at	closed cycle
at (atm/°C)	gas turbine	-	102/538	gas turbine
Max Surface Temp (°C)	870	815	724	1,000
Max Fuel Temp (°C)	880	1040	1052	1350
Coolant	He	He	He	He
Coolant mass flow (Kg/sec)	35.5	6.3	30	40.6
Coolant inlet temp (°C)	410	400	290	425
Coolant outlet temp (°C)	705	700	650	735
Coolant pressure (atm)	68	75	56.5	25
Moderator	graphite	BeO	water	graphite
Fuel	UO_2/ss	UO_2-BeO	UO_2	UO_2/ThO_2
Fuel Form	tube	Rod	Rod	coated part.
Fuel enrichment (%)	93	62.5	6	90
cladding	316 ss	Hastelloy X	Incoloy-800	Pyro C
cladding thickness (mm)	.25	.5	.4	-
Control material	Eu_2O_3	*	B_4C	B_4C
Efficiency (%)	31	-	33.7	39

Table 6-4

Technical Data of Related Mobile Reactors

Reactor	ML-1[45]	ANP[38]	NERVA XEP[46]	NERVA NTR[41]
Year activated	1963	1972	1964	1971
Power (MWt)	3.3	300	1126	2.2
Power (MWe)	.33	*	-	.506
Active Core Dia (m)	.56	*	*	.52
Active Core Length (m)	.56	*	*	.52
Avg Power Density (KW/l)	16.3	*	*	26.8
Power Conversion at (atm/°C)	closed cycle gas turbine	close/open gas turbine	None Thrust	closed cycle gas turbine
Max Surface Temp (°C)	955	1150	*	920
Max Fuel Temp (C)	1450	*	*	1120
Coolant	N_2	He	H_2	He/Xe
Coolant Mass Flow (Kg/sec)	11.3	*	32	17.5
Coolant Inlet Temp (°C)	422	538	- 144	627
Coolant Outlet Temp (°C)	650	945	2000	872
Coolant Pressure (atm)	20.5	102	38	15.5
Moderator	water	water	graphite	graphite
Fuel	UO_2/BeO	UO_2	UC_2	UC_2
Fuel Form	Rod	Pin	coated part.	coated part.
Cladding	Hastelloy X	TZM	Pyro-C	Pyro-C
Cladding Thickness (mm)	.75	.25	-	-
Control Material	Ag Cd In	Boron Alloy	Copper Boron	Copper Boron
Efficiency (%)	10	*	-	23

VII. CONCLUSIONS

The present status of maritime gas cooled reactors has been reviewed in the previous pages. As has been shown, all developmental projects have been cancelled before fruitation and no new projects are being accepted. Therefore, the prospects of seeing a nuclear merchant ship powered by a gas cooled reactor in the relatively near future are slim. Furthermore, a reasonable estimate of the present cost of developing such a system is probably 500 million dollars. This high cost could only be justified in a clear-cut case of an existing need. Such is not the case. Unless there is a demand for very high endurance with a low weight to horsepower ratio, the chemical power plant, with its promise of continuing improvement and its past history of reliability, will continue to be the system desired.

The gas cooled nuclear power plant even if it fulfilled its promise of good thermal efficiency, low weight to horsepower ratio, and fuel economy, would still be restricted in its installation because of the hazards to personnel, hazards in case of crash, problems of control and accessibility, and, probably more to the point, the political environment. As indicated by the fate of the N. S. SAVANNAH, there is more to success than technological feasibility and desirability. The political environment and proper timing have perhaps more to do with technological success or failure than engineering problems or solutions. A ship must be able to use the world's harbors without too many restrictions, which will not be the case in the near future for

nuclear ships. A civilian ship must also contend with the demands of labor, which for nuclear ships can cripple technical effectiveness.

The end result is that the demand must be so great that the rewards of technical development and its political acceptance are really worth it. If such a demand does come about, perhaps due to today's energy crisis, or a broad need for high endurance craft such as the development of in-ocean cities, the gas cooled reactor as epitomized by the designs discussed in this paper will need to be developed. It appears to the author that the present state of nuclear rocket technology with its huge capital and scientific investment and relative success should logically be the starting point for future development. It has inherent a low weight to horsepower ratio, small physical size, ruggedness, and a tentatively proven reliability.

LIST OF REFERENCES

1. "Parametric Design Considerations for Nuclear Applications to Surface Effect Vehicles," James L. Decker, Joint Surface Effects Ships Program Office, Washington, D.C., July 1970.

2. "Design Considerations for Nuclear Applications to Surface Effect Vehicles," S. Eldridge, Aerojet-General Corporation, July 1970.

3. "GCR-AGR-HTGR - Development of the Gas Cooled Reactor Concept," H. Kramer, Kerntechnik 14(10):439, October 1972.

4. "Gas-Cooled Clad-Fuel Reactors," Walter Mitchell III, Power Reactor Technology 8(1):58, Winter 1965.

5. "Safety Features of an Advanced Gas Cooled Reactor," A. Schatz and H. Kley, Kerntechnik 11(7):371, July 1969.

6. "Final Hazards Summary Report, Peach Bottom Atomic Power Station #1," Philadelphia Electric Company, Philadelphia.

7. "Gas Cooled Reactors" Power Reactor Technology 5(3):60, Summer 1962.

8. "Coated Particle Fuels in High Temperature Reactors: A Summary of Current Applications," W. V. Goeddel, Nuclear Applications 3(10):599, October 1967.

9. "Gas Cooled Reactors - The Dragon Reactor," Power Reactor Technology, 6(1):74, December 1962.

10. "Fort St. Vrain Nuclear Generating Station, Preliminary Safety Analysis Report," Public Service Company of Colorado, Denver, September 1966.

11. "Core Design Characteristics," R. C. Dahlberg, R. F. Turner and W. V. Goedell, Nuclear Engineering International 14(63):1073, December 1969.

12. "The 300 MW THTR Nuclear Power Station at Uentrop," H. W. Muller, Kertechnik 14(9):385, September 1972.

13. "UHTREX: Alive and Running with Coolant at 2400°F," Nuclear News 12(11):31, November 1969.

14. "UHTREX Operation Near," B. M. Hoglund, Reactor Technology 9(1):51, Winter 1965-1966.

15. "Operating Experience and Design Trends for Helium Cooled Nuclear Power Reactors," ORNL-TM-2946, D. B. Trauger, Oak Ridge National Laboratory, Tennessee, June 1970.

16. "Maritime Reactors," Power Reactor Technology 8(1):97, Winter 1964-1965.

17. "Current Status and Future Prospects of Nuclear Ship Propulsion Systems," Kurt Illies, Kerntechnik 15(3):114, March 1973.

18. "Design Practice: The N. S. SAVANNAH," Power Reactor Technology 6(1):43, December 1962.

19. "Nuclear Propulsion for Merchant Ships," A. W. Kramer, U. S. Government Printing Office, Washington, D.C., 1962.

20. "Some Operational Experiences with the Nuclear Powered Bulk Carrier 'Otto Hahn'," The Motorship 51:387, November 1970.

21. "Safety Design Aspects of Nuclear Powered Ship Otto Hahn," W. Wiebe, D. Ulken, C. Salander, and K. Henning, Nuclear Safety 9(6):443, November/December 1968.

22. "Atomshiff, Zan Than in Betrieb," Atomwirtsdraft 17(9/10):444, September/October 1972.

23. "Study Contract MA 1252 for 20,000 SHP Nuclear Propulsion System for Tanker," 3 Volumes, Research Staff, General Motors Corporation, Detroit, April 1957.

24. "Design Objectives for the Maritime Gas Cooled Reactor Land Based Prototype," GAMD-1722, J. D. Messersmith, W. C. Moore, and H. N. Wellhouser, General Atomic Division, General Dynamics Corp., San Diego, 1962.

25. "A Review of the Maritime Gas Cooled Reactor Program," GA-2603, Kenneth A. Trickett, General Atomic Division, General Dynamics Corp., San Diego, 1961.

26. "Maritime Gas Cooled Reactor Project Engineering Report No. EC-167 on Development of Turbomachinery (Summary Report)," Westinghouse Electric Corporation, Small Steam and Gas Turbine Division, Lester, Pennsylvania, June 1963.

27. "Experimental Beryllium Oxide Reactor Program, Quarterly Progress Report," GA 7485, General Atomic Division, General Dynamics Corp., San Diego, February 8, 1967.

28. "630A Mark V Maritime Nuclear Steam Generator Status Report," GEMP-342, General Electric Company, Cincinnati, Ohio, February 1965.

29. "Design of a Marine Nuclear Power Plant Utilizing the Direct Brayton Cycle," Bruce S. Gathy, Iowa State University, 1967.

30. "The 25-MW(e) Geesthacht KSH Nuclear Power Plant," E. Bohm, A. Ehret, H. Geppart, W. Hauck and A. Klusmann, Symposium on Advanced and High Temperature Gas Cooled Reactors, (STI/PUB/197) Julich, October 1968, p. 109.

31. "Gas Turbine Nuclear Power Plants," A. A. Katterhenry, Reactor Technology, 13(1):7, Winter 1969-1970.

32. "High Temperature Reactors with Helium Turbines as a New Concept for Nuclear Power Plants," W. Kliefoth, Kerntechnik 11(2):65, February 1969.

33. "High Temperature Reactors with Gas Turbines," E. Bohm, Atomkeinenergie 11(9/10):343, September/October 1966.

34. U. S. Maritime Administration, private communication.

35. "Army Gas Cooled Reactor Systems Program, Final Report," IDO-28655, Aerojet-General Nucleonics, San Ramon, California, March 1966.

36. "Gas Cooled Reactors-Army Reactor Systems Program," Power Reactor Technology 6(3):57 (1963).

37. "Status of the Nuclear Powered Airplane," Frank E. Rom, Journal of Aircraft 8(1):26.

38. "A Study Program for an Advanced Plant for Nuclear Merchant Ship Propulsion," Westinghouse Electric Corporation, Unpublished internal document.

39. "Nuclear Thermal Rockets," J. E. Morrissey, Kerntechnik 11(3):152, March 1969.

40. "The Nuclear Rocket Program - Its Status and Plans," L. C. Corrington, Journal of Spacecraft and Rockets 6(4):465, April 1969.

41. "Nerva Technology Reactor Integrated with NASA Lewis Brayton Cycle Space Power System," L. H. Boman and J. G. Gallagher, Journal of Spacecraft and Rockets, 8(5):500, May 1971.

42. "Gas Core Reactor Technology," J. D. Clement and J. R. Williams, Reactor Technology 13(3):226, Summer 1970.

43. "Review of Fission Engine Concepts," K. Thom, Journal of Spacecraft and Rockets, 9(9):633, September 1972.

44. "Power Reactors 73," Nuclear Engineering International, 18(203): 325, April 1973.

45. "Thermal Design Aspects of Gas Cooled Power Reactor Cores," A/CONF 28/P/226, M. Toost, General Atomic Division, General Dynamics Corporation, San Diego.

46. "Operational Characteristics of Nuclear Rockets," David Buden, Journal of Spacecraft and Rockets 7(7):832, July 1970.

47. "Proceedings of the Gas Cooled Reactor Information Meeting at the Oak Ridge National Laboratory, April 27-30, 1970," CONF-700401, Oak Ridge National Laboratory, Tennessee.

APPENDIX A

GAS AND VAPOR POWER CYCLES

A brief review of power cycles is presented based upon the information contained in Thermodynamics by Kenneth Wark, McGraw-Hill, Second Edition, 1971.

The conversion of heat into a more useful form of energy, work, is usually accomplished by a cyclic device. These cyclic processes should be such that the thermal efficiency - the ratio of net work output to heat input - is a maximum. A heat engine is any system which executes a cycle while receiving heat and producing some net work output. The maximum thermal efficiency of any heat engine operating between two constant temperature reservoirs is given by the Carnot efficiency -

$$\eta_{Carnot} = \frac{T_H - T_L}{T_H} \tag{A-1}$$

where η_{Carnot} = Carnot thermal efficiency

T_H = temperature of the hot reservoir in degrees absolute

T_L = temperature of the cool reservoir in degrees absolute

The efficiency of any heat engine operating between two constant-temperature reservoirs is increased by either employing supply reservoirs of very high temperatures or rejecting to heat reservoirs of extremely low temperature. The latter case is difficult to carry out

since temperatures below ambient conditions must be artificially acquired. Usually materials limitations restrict operating at higher temperatures but this is the area where most of the development is occurring in attempts to increase thermal efficiencies of power conversion equipment. When one considers that practical engines may have a heat supply, the temperature of which varies considerably below the maximum value and that mechanical efficiencies must also be included, it is not too surprising that modern heat engines have overall efficiencies of only 25 to 40 percent. It is essential to understand that even under optimum theoretical conditions, a heat engine is relatively inefficient in converting heat to work by means of a cyclic process.

CARNOT CYCLE

Equation A-1 pertains to theoretical heat engines which operate in a totally reversible manner between fixed temperature reservoirs. One of the best known of these is one which operates under conditions called a Carnot cycle. Basically, the Carnot cycle requires that the working medium of the engine undergo four processes which constitute a cycle. Figure A-1 shows a temperature-entropy (T-s) diagram for a Carnot cycle. As can be seen, the Carnot cycle is comprised of two isothermal externally reversible processes and two reversible adiabatic processes. In process 2-3 heat is added to the working medium and the working medium of the engine is allowed to expand isothermally producing a work output. From state 3, the working medium is allowed to expand reversibly and adiabatically to state 4. During process 3-4

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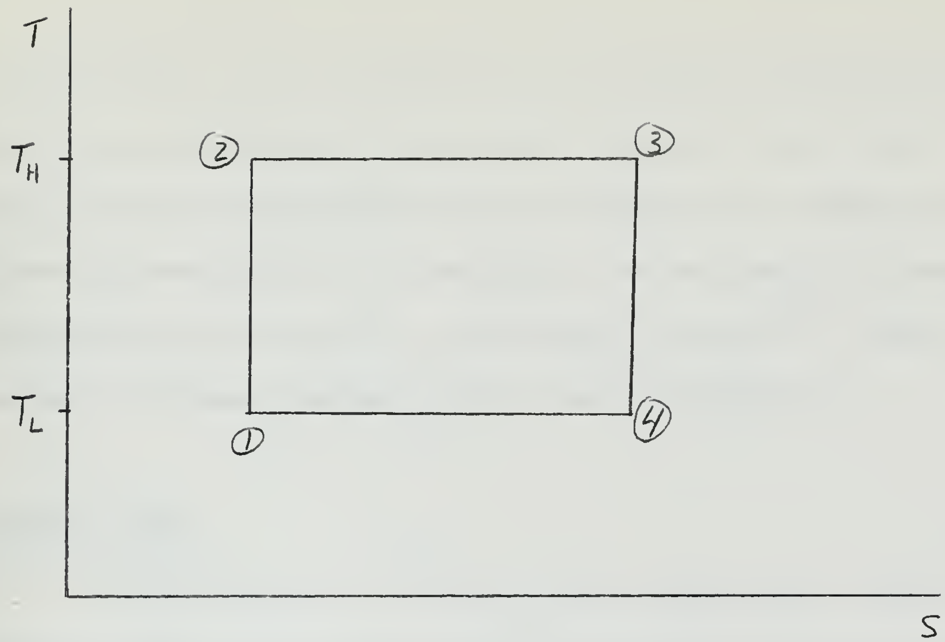


Figure A-1

T-s Diagram for a Carnot Cycle

additional work is produced. The system is now compressed isothermally to state 1. During the compression process, heat is rejected to a reservoir at a temperature T_L. State 1 is selected so that, by a final reversible, adiabiatic compression, the working medium is returned to its initial state via process 1-2.

The heat supplied to and rejected from a Carnot engine is easily seen in Figure A-1. For an isothermal reversible process heat equals the temperature times the change in entropy. Therefore, the area beneath the horizontal line 2-3 represents the heat added and the area beneath the horizontal line 4-1 represents the heat rejected. From an

energy balance on the heat engine the difference between the heat added and the heat rejected is the net work produced by the engine during the cycle. This is represented by the enclosed area in Figure A-1. By either increasing T_H or decreasing T_L, the enclosed area becomes a larger fraction of the total area beneath the line 2-3, hence the thermal efficiency increases under these circumstances.

RANKINE CYCLE

A modification of the Carnot cycle which uses a vapor as its working fluid is the Rankine cycle. It is the basic cycle for all power conversion systems employing steam as the working fluid. In its most basic form it is shown in Figure A-2. It consists of an isentropic compression in a pump (1-2), constant pressure heat addition in a boiler or heat exchanger (2-2'-3), isentropic expansion in a turbine (3-4), and constant pressure heat removal in a condenser (4-1). As in the Carnot cycle, the net work is represented by the area enclosed by the cycle. The efficiency of the Rankine cycle can be increased by doing three things: 1) superheating, 2) reheating, and 3) regeneration. A diagram including all three is shown in Figure A-3.

The process of superheating (the temperature difference between 2' and 3) leads to a higher temperature at the turbine inlet without increasing the maximum pressure in the cycle. This increases the average temperature of the heat addition process which increases the efficiency on a Carnot engine analysis and also increases the quality of the steam in the turbine which increases its mechanical efficiency.

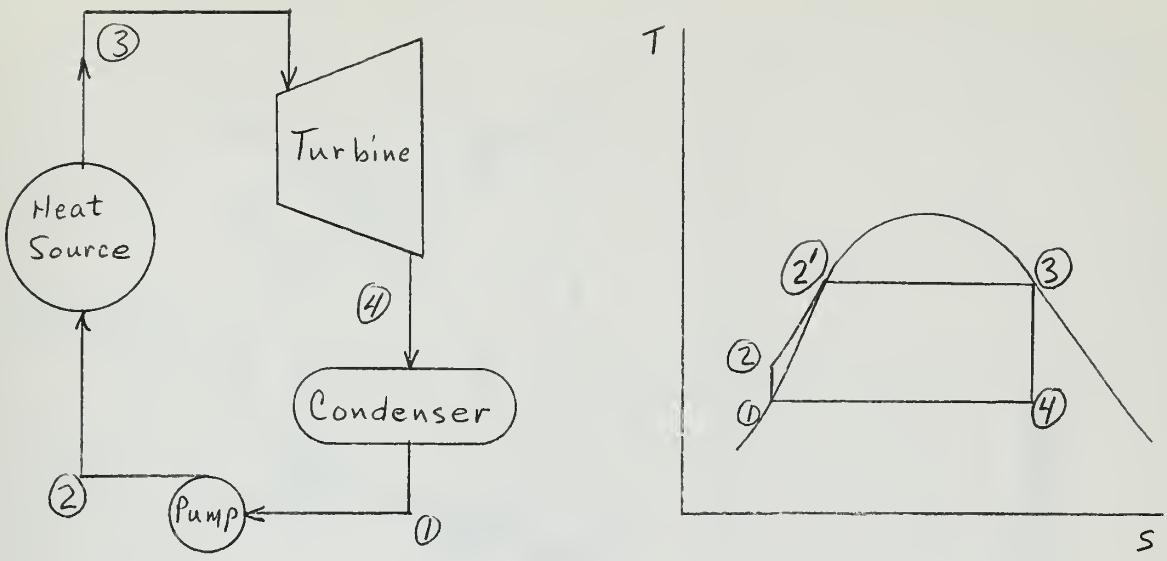


Figure A-2

A Simple Power Conversion System Operating
on the Rankine Cycle

To decrease the pressure, but maintain the same average temperature of the heat addition process, the steam is not allowed to expand completely to the condenser pressure in one stage. The partially expanded steam in state 4 is sent back to the boiler for reheating to state 5 and allowed to further expand in the low pressure turbine. This also, improves the quality of the steam in the turbines and therefore increases their mechanical efficiency. The third method of increasing the average temperature of the heat addition process is to utilize a portion of the expanding steam in state 6 to heat the feedwater coming from the condenser in state 9 before it goes into the boiler. They mix to produce state 1 in the cycle shown. More complicated schemes

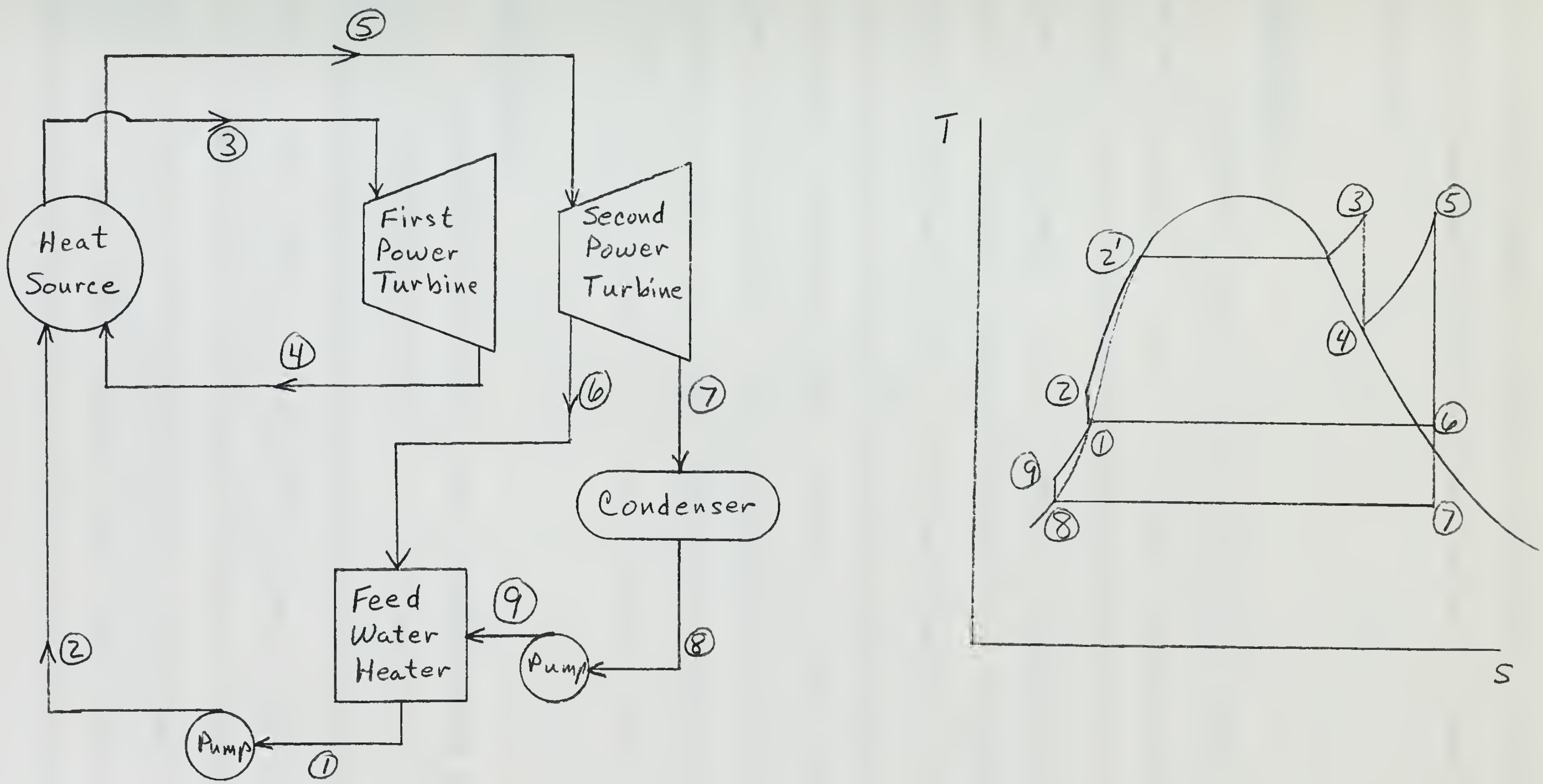


Figure A-3

Ideal Regenerative, Reheat Rankine Cycle with Superheat
and One Open Feedwater Heater

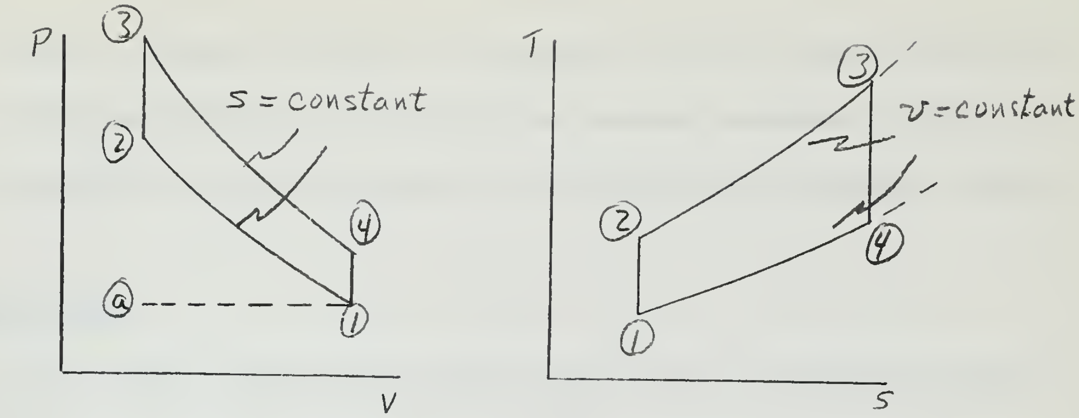
utilizing many stages of expansion and feedwater heating are utilized in practice, but they all have the same purpose, to raise the overall efficiency.

OTTO CYCLE

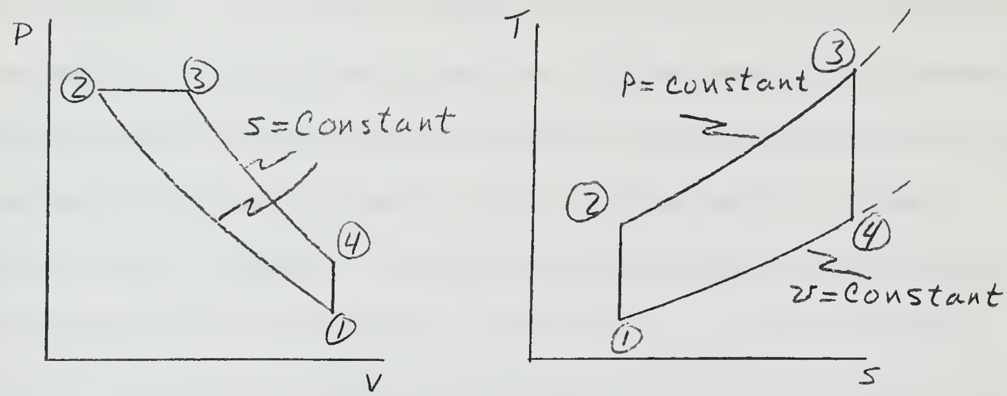
A different class of cycles using gases rather than vapors for the working medium are used for internal combustion engines and gas turbines. A four stroke Otto cycle is composed of four internally reversible processes, plus an intake and an exhaust portion of the cycle. It is shown at the top of Figure A-4. It consists of an adiabatic compression (1-2), a constant volume heat addition (2-3), an adiabatic expansion (3-4), and a constant volume heat rejection (4-1). In addition, an exhaust stroke (1-a) and an intake stroke (a-1) are shown for completeness. This is the principle cycle used in internal combustion engines. The thermal efficiency, neglecting changes in specific heats, is a function of the compression ratio.

DIESEL CYCLE

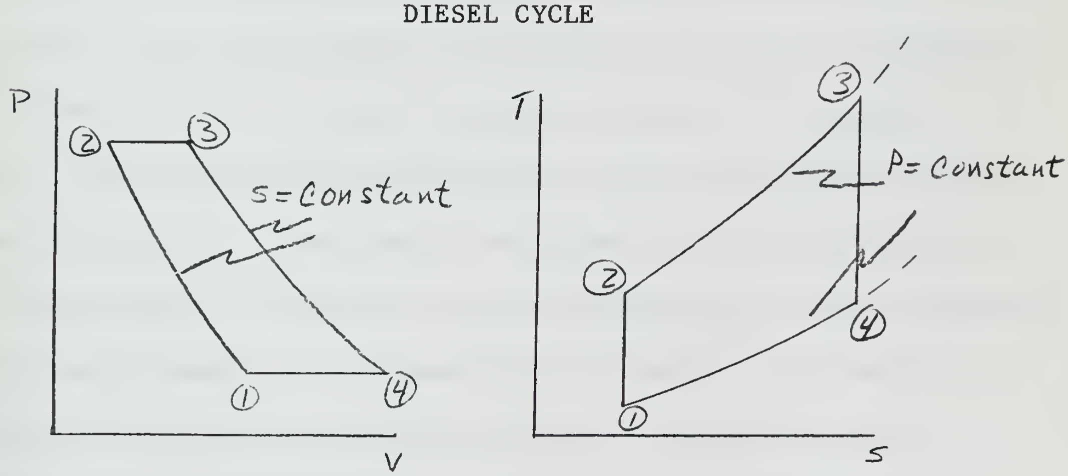
The Diesel cycle is also used in reciprocating engines and is shown in the middle of Figure A-4. It is similar to the Otto cycle except that the process 2-3 is a constant pressure heat addition. In actual practice this is obtained by spraying the fuel into the cylinder at a controlled rate. The thermal efficiency of the Diesel cycle is also a function of the compression ratio, neglecting variable specific heat values; plus it is a function of the cutoff ratio which relates the increase in volume during the constant pressure process (2-3) to



OTTO CYCLE



DIESEL CYCLE



BRAYTON CYCLE

Figure A-4

The Basic Ideal Gas Power Cycles

the original volume, at state 2. This means that the Diesel cycle has lower efficiency than the Otto cycle for the same compression ratio, but in practice is usually operated at a much higher compression ratio.

BRAYTON CYCLE

In a simple gas turbine power cycle, separate equipment similar to that used in vapor cycles is used for the various processes of the cycle. Initially, the working medium is compressed adiabatically in a rotating axial or centrifugal compressor. At the end of this process the working medium enters a heat exchanger or other source of heat such as a combustion chamber or reactor core. The heated gas is then expanded through a turbine. A cycle composed of these three steps is called an open cycle since the gas, usually air, is sucked into the compressor from the environment and exhausted from the turbine to the environment. In a closed cycle, the fourth process of heat rejection must be carried out to return the working medium to the initial temperature. This cycle is called a Brayton cycle and is shown at the bottom of Figure A-4. The thermal efficiency of the basic Brayton cycle is primarily a function of the overall pressure ratio, neglecting kinetic energy changes and variable specific heats. Increasing the allowable turbine inlet temperature allows one to increase the pressure ratio, and therefore increase the thermal efficiency.

It should be kept in mind that because of fluid friction and heat losses, irreversibilities occur in the cycle. The effect of these irreversibilities in the compressor and the turbine are usually accounted for by defining an efficiency for them called adiabatic

efficiency, which for a turbine is equal to the ratio of the actual work out to the isentropic work out, and for a compressor is the ratio of isentropic work in to actual work in. Efficiencies greater than 80 percent are not uncommon in modern turbomachinery, but the effect of the irreversibilities is to require a larger compressor work input which is available from a smaller turbine output. Hence the compressor in actual practice may consume 40 to 70 percent of the turbine output.

The basic gas turbine cycle can be modified in several important ways beyond increasing the turbine inlet temperature to increase its overall efficiency. One of these is the concept of regeneration. If the outlet turbine temperature is higher than the outlet compressor temperature, it is possible to reduce the amount of heat added externally by reheating the gas leaving the compressor with energy taken from the turbine exhaust gas. The exchange of heat between the two flow streams takes place in a heat exchanger usually called a regenerator, or recuperator.

Another method of increasing the overall efficiency of the gas turbine cycle is to decrease the work input to the compression process and/or increase the work output of the turbine. The effect of either of these procedures is to increase the work output of the engine. A method of decreasing the work input to the compression process is to cool the gas during compression. In many cases it is either not possible or practical to have much heat transfer through the compressor casing. To achieve the benefits of cooling, multistage compression with intercooling is frequently used. In this process the gas is first compressed to some intermediate pressure and then passed through an

intercooler (a heat exchanger), where it is cooled down at essentially constant pressure. Then it passes through another stage of compression, where its pressure is increased further. This would be followed by another intercooler process and then another stage of compression, etc., until the final high pressure is reached. The overall result is lowering the net work required for a given pressure ratio.

In conjunction with intercooling it is often found effective to also use turbine staging with reheat between stages. However, with a reactor as the heat source, this so complicates the process that it is rarely done. However, turbine staging may be done for another reason. If the compressor speed and the required output shaft speed are very different, often two stages of turbine are employed, one to run the compressor and one to produce output. A cycle which employs regeneration, intercooling, a low pressure output turbine, and a high pressure compressor turbine is presented in Figure A-5. Using a reactor as a heat source this is typical of a maritime closed cycle gas turbine power conversion cycle.

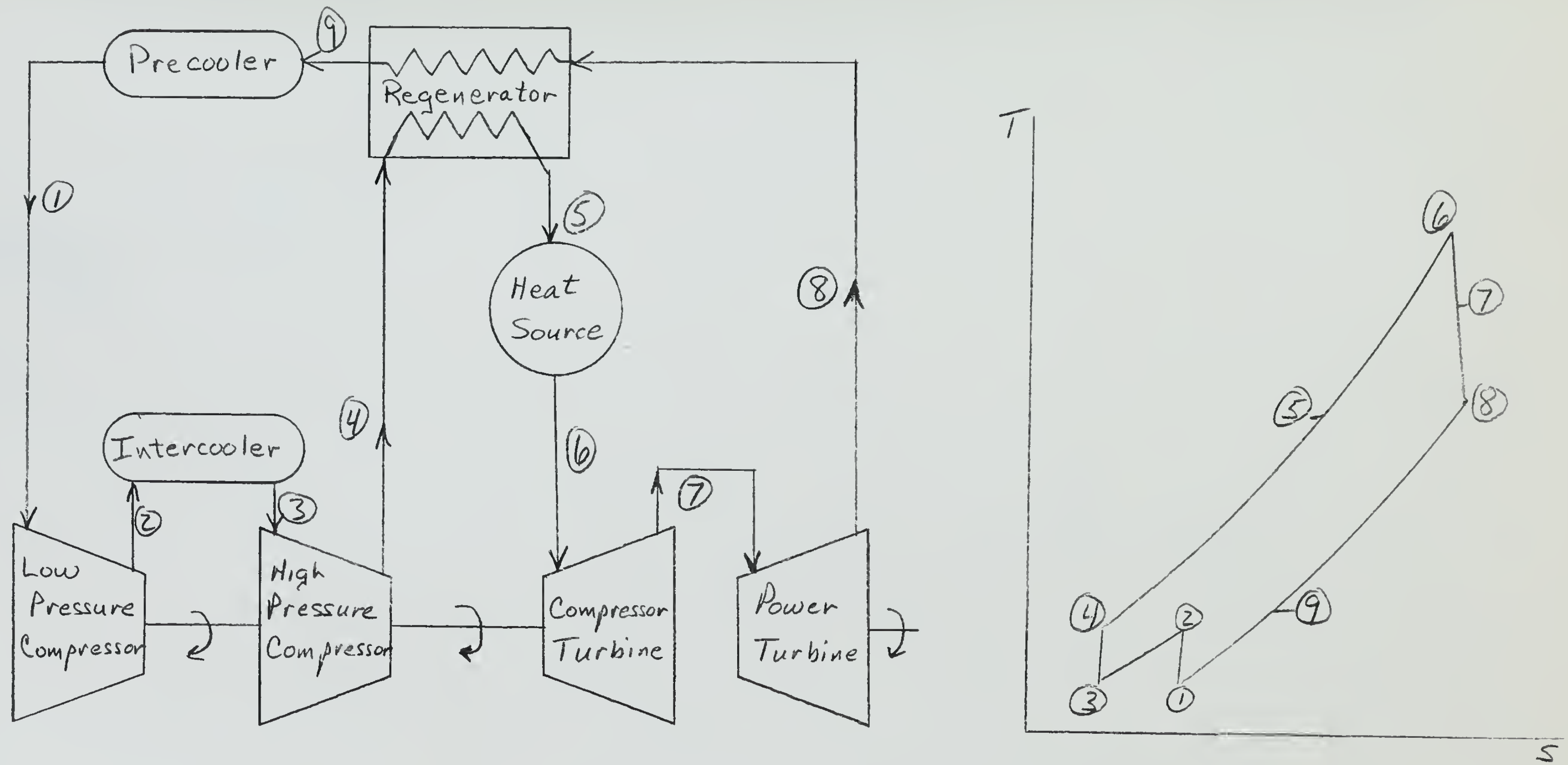


Figure A-5

Gas Turbine Cycle with Regeneration, Intercooling, and Two Turbines